CONVENTION ON NUCLEAR SAFETY

NATIONAL NUCLEAR SAFETY REPORT

ARGENTINA - 2019

Autoridad Regulatoria Nuclear
PRESIDENCIA DE LA NACIÓN

EIGHT REPORT

ARGENTINEAN NATIONAL REPORT FOR THE CONVENTION ON NUCLEAR SAFETY

August **2019**

Autoridad Regulatoria Nuclear
PRESIDENCIA DE LA NACIÓN

This report demonstrates how Argentina has implemented its obligations under the Convention on Nuclear Safety. The report follows closely the guidelines, regarding form and structure, that were established by the contracting parties under Article 22 of the Convention.

This Report is produced by the Autoridad Regulatoria Nuclear (Nuclear Regulatory Authority) on behalf of Argentina. Contributions to the report were made by representatives from Nucleoeléctrica Argentina S.A. (NA-SA) and Comisión Nacional de Energía Atómica (CNEA)

ARGENTINEAN NATIONAL REPORT
FOR THE CONVENTION ON NUCLEAR SAFETY
Eight Report

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GLOSSARY

ABACC	*Agencia Brasileño-Argentina de Contabilidad y Control de Materiales Nucleares* (Argentine-Brazilian Agency for Accounting and Control of Nuclear Materials)
AEA	Atomic Energy Agency (International)
AECL	Atomic Energy of Canada Limited
AGE	*Área de Gestión Ezeiza* (Ezeiza Waste Management Area)
AMP	Ageing Management Program
ANL	Argonne National Laboratory
ANSeS	*Administración Nacional de la Seguridad Social* (National Social Security Administration)
AOO	Anticipated Operational Occurrences
AREVA	AREVA NP
ARN	*Autoridad Regulatoria Nuclear* (Nuclear Regulatory Authority - Argentina)
ASECQ	*Almacenamiento en Seco de Elementos Combustibles Quemados* (Dry Storage of Spent Fuel Elements)
ATWS	Anticipated Transients Without Scram
BDBA	Beyond Design Basis Accident
BP	Break Preclusion
B&W	Babcock & Wilcox
CA	Condition Assessment
CALPIR	*Consejo Asesor para el Licenciamiento de Personal de Instalaciones Relevantes*
CANDU	Canada Deuterium Uranium
CAREM	*Central Argentina de Elementos Modulares* (Argentinean Nuclear Power Plant of Modular Elements)
CCE	*Centro de Control de Emergencias* (Emergency Control Center)
CDF	Core Damage Frequency
CDS	Core Damage States
CECE	*Centro Externo de Control de Emergencias* (External Emergency Control Center)
CHF	Critical Heat Flux
CIAS	*Comité Interno Asesor de Seguridad* (Internal Safety Advisory Commitee)
CICE	*Centro Interno de Control de Emergencias* (Internal Emergency Control Center)
CIP	*Centro de Información Pública* (Public Information Center)
CISIN	*Control e Inspección de Seguridad de Centrales* Nucleares (NPPs Safety Control and Inspection)
CNA I	*Central Nuclear Atucha I* (Atucha I Nuclear Power Plant)
CNA II	*Central Nuclear Atucha II* (Atucha II Nuclear Power Plant)
CNE	*Central Nuclear Embalse* (Embalse Nuclear Power Plant)
CNEA	*Comisión Nacional de Energía Atómica* (National Atomic Energy Commission)
CNNC	China National Nuclear Corporation
CNS	Convention on Nuclear Safety
COEM	*Centro Operativo de Emergencias Municipal* (Municipal Emergency Operative Center)
COG	CANDU Owner´s Group
CONICET	*Consejo Nacional de Investigaciones Científicas y Técnicas*
CONUAR	*Combustibles Nucleares Argentinos S.A.*
CPR	Corporate Peer Review
CRMRD	*Comité de Revisión de Modificaciones Relevantes al Diseño* (Relevant Design Changes Review Committee)

CRT	*Comité de Revisión Técnica* (Technical Review Commitee)
CSS	Commission on Safety Standards
DA	Design Authority
DBA	Design Basis Accidents
DBE	Design Basis Earthquake
DCC	Digital Control Computers
DEC	Design Extension Conditions
DG	Diesel Generator
DID	Defence in Depth
DNBR	Departure from Nucleate Boiling Ratio
DOE	U.S. Department of Energy
DSA	Deterministic Safety Assessment
DNPC	*Dirección Nacional de Protección Civil* (National Civil Protection Direction)
DOE	U.S. Department of Energy
ECCS	Emergency Core Cooling System
EduTA	Education and Training Appraisal
EECC	External Emergency Control Centre
EERO	External Emergency Response Organization
EMS	Environmental Management System
ENACE	*Empresa Nuclear Argentina de Centrales Eléctricas* (Argentinean Nuclear Power Plants Company)
ENREN	*Ente Nacional Regulador Nuclear* (Nuclear National Regulatory Body) [predecessor of ARN]
ENSI	*Empresa Neuquina de Servicios de Ingeniería S.E*. (Neuquen Engineering Services Company)
EOL	End of Life
EPRI	U.S. Electric Power Research Institute
EPS	Emergency Power Supply
EPZ	Extended Complementary Planning Zone
EQ	Environmental Qualifications
ERO	Emergency Response Organization
EWS	Emergency Water Supply
FAE	*Fabricación de Aleaciones Especiales*
FAMOS	Fatigue Monitoring System
FCVS	Filtered Containment Venting System
FE	Finite elements
FIUBA	*Facultad de Ingeniería de la Universidad de Buenos Aires*
FORO	*Foro Iberoamericano de Organismos Reguladores Radiológicos y Nucleares* (Ibero-American Forum of Radiological and Nuclear Regulatory Agencies)
FPY	Full Power Years
FSAR	Final Safety Analysis Report
GIS	Geographic Information System
GRS	*Gesellschaft für Anlagen-und Reaktorsicherheit* (Reactor and Facilities Safety Corporation)
GRS	Ground Response Spectra
GSR	General Safety Requirements
HA	Human Actions
HCLPF	High Confidence of Low Probability of Failure

HEU	Highly Enriched Uranium
HPES	Human Performance Evaluation System
HPIP	Human Performance Improving Program
HPIP	Human Performance Investigation Program
HRA	Human Reliability Analysis
HXs	Heat Exchangers
IAEA	International Atomic Energy Agency
IAMP	Integrated Ageing Management Program
ICR	Inappropriate Condition Report
ICRP	International Commission on Radiological Protection
IDIA	*Instituto de Investigaciones Antisísmicas "Ing. Aldo Bruschi"*
IECC	Internal Emergency Control Centre
IFMAP	Irradiated Fuel Management Advisory Programme
IMPSA	*Industrias Metalúrgicas Pescarmona S.A.*
INEEL	Idaho National Engineering and Environmental Laboratory
INES	International Nuclear Event Scale
INPRES	*Instituto Nacional de Prevención Sismica*
INPRO	International Project on Innovative Nuclear Reactors and Fuel Cycles
INSAG	International Nuclear Safety Group
INVAP	INVAP S.E.
IPERS	International Peer Review Service
IRAM	*Instituto Argentino de Normalización y Certificación (ex Instituto de Racionalización Argentino de Materiales)*
IRRS	Integrated Regulatory Review Service
IRS	Incident Reporting System
ISI	In Service Inspection
IST	Industry Standard Toolset
IXP	International Exchange Program
JOEN	*Jefe Operativo de Emergencias Nucleares* (Nuclear Emergency Operative Chief)
KWU	Kraftwerk Union
LAC	Local Air Coolers
LBB	Leak Before Break
LCNRD	Licensing and Control of Nuclear Reactors Department
LISS	Liquid Injection Shutdown System
LOCA	Loss of Coolant Accident
LOFA	Loss of Flow Accident
LOOP	Loss of Offsite Power
LPD	Linear Power Density
LTA	Long Term Agreement
LTO	Long Term Operation
MCCI	Molten Core-Concrete Interaction
MCR	Main Control Room
MDG	Mobile Diesel Generator
MOU	Memorandum of Understanding
MSSV	Main Steam Safety Valves
NA-SA	*Nucleoeléctrica Argentina S.A.*
NDE	Non Destructive Examinations
NNSA	National Nuclear Safety Administration (China)

NNSA	National Nuclear Security Administration (U.S)
NPP	Nuclear Power Plant
NR	National Report
NSGC	Nuclear Security Guidelines Committee
NUSSC	Nuclear Safety Standards Committee
OAA	*Organismo Argentino de Acreditación*
OBE	Operating Basis Earthquake
OEP	Operating Experience Program
OJT	On-the-Job Training
OPDS	*Organismo Provincial para el Desarrollo Sostenible*
OPEX	Operating Experience
ORE	*Organización de Respuesta ante Emergencias* (Emergency Response Organization)
ORNL	Oak Ridge National Laboratory
OSART	Operational Safety Review Team
PARs	Passive Auto-catalytic Recombiners
PAZ	Precautionary Action Zone
PGA	Peak Ground Acceleration
PHTS	Primary Heat Transport System
PHWR	Pressurized Heavy Water Reactor
PIEs	Postulated Initiating Events
PLEX	Plant Life Extension
PLiM	Plant Life Management
PMH	Probable Maximum High Water Level
POEAs	*Procedimientos Operacionales para Eventos Anormales* (Operating Procedures for Abnormal Events)
PPS	Physical Protection System
PR	Peer Review
PRACS	*Programa de Afianzamiento de la Cultura de la Seguridad* (Programme of Consolidation of Safety Culture)
PSA	Probabilistic Safety Assessment
PSHA	Probabilistic Safety Hazard Analysis
PSR	Periodic Safety Review
PTS	Pressure Thermal Shock
PWR	Pressurized Water Reactor
QA	Quality Assurance
QMS	Quality Management System
RASSC	Radiation Safety Standards Committee
RB	Reactor Building
RLE	Review Level Earthquake
RPV	Reactor Pressure Vessel
RSMC	Regional Specialized Meteorological Center
RTC	Regional Training Center
RWO	Risk Work Order
SACRGs	Severe Accident Control Room Guidelines
SAEGs	Severe Accident Exit Guidelines
SFP	Spent Fuel Pool
SAGs	Severe Accident Guidelines
SAGSI	Standing Advisory Group on Safeguards Implementation

SALTO	Safety Aspects of Long Term Operation
SAMG	Severe Accident Management Guidelines
SAMP	Severe Accident Management Program
SANDIA	Sandia National Laboratories
SAR	Safety Analysis Report
SARIS	Self – Assessment of the Regulatory Infrastructure for Safety
SAT	Systematic Approach to Training
SBO	Station Black Out
SC	Safety Culture
SCD	Severe Core Damage
SCGs	Severe Challenge Guidelines
SCK/CEN	Studiecentrum voor Kernenergie (Belgian Nuclear Research Center)
SCR	Secondary Control Room
SDS	Shut – down Systems
SEDA	*Sistema de Evaluación de Dosis en Emergencia* (Accidental Dose Assessment System)
SF	Spent Fuel
SG	Steam Generator
SHS	Secondary Heat Sink
SIEMENS	SIEMENS Kraftwerk Union AG
SIEN	*Sistema de Intervención en Emergencias Nucleares* (Nuclear Emergency Response System)
SIER	*Sistema de Intervención en Emergencias Radiológicas* (Radiological Emergency Response System)
SIFEM	*Sistema Federal de Emergencias* (Federal Emergency System)
SINAGIR	*Sistema Nacional para la Gestión Integral del Riesgo y la Protección Civil* (National System for Comprehensive Risk Management and Civil Protection)
SL	Seismic Level
SMA	Seismic Margin Assessment
SMN	*Servicio Meteorológico Nacional* (National Meteorological Service)
SOP	Strategic Observation Program
SPI	Safety Performance Indicators
SSA	Safety Seismic Assessment
SSC	Structures, Systems and Components
SSCRS	Structures, Systems and Components Relevant to Safety
SSEL	Safe Shutdown Equipment List
TLAA	Time Limited Ageing Analysis
TM	Technical Meeting
TRANSSC	Transport Safety Standards Committee
TS	Technical Support
TSO	Technical Support Organizations
TSP	Tube Support Plate
TÜV	*Technischer Überwachungs Verein, Baden* (German Inspection Organization)
UBA	*Universidad de Buenos Aires* (Buenos Aires University)
UCE	*Unidad de Capacitación y Entrenamiento* (Education and Training Unit*)*
UHRS	Uniform Hazard Response Spectra
UNIPI	University of Pisa
UNSCEAR	United Nations Scientific Committee on the Effects of Atomic Radiation

UNSJ	*Universidad Nacional de San Juan* (National University of San Juan)
UPS	Uninterruptible Power Supply
UPZ	Urgent Protective Action Planning Zone
USNRC	US Nuclear Regulatory Commission
UTN-BA	*Universidad Tecnológica Nacional - Buenos Aires* (National Technological University)
VDNS	Viena Declaration on Nuclear Safety
WANO	World Association of Nuclear Operators
WASSC	Waste Safety Standards Committee
WENRA	Western European Nuclear Regulators Association
WHO	World Health Organization
WMO	World Meteorological Organization
WP	Work Permit

INTRODUCTION

This *Eight* National Nuclear Safety Report is an updated report that includes all safety aspects of the Argentinean nuclear power plants (NPPs) and the measures taken to enhance their safety.

In addition, the recommendations stated by the President for CNS 8th Review Meeting with regard the Vienna Declaration on Nuclear Safety have been taken into consideration as well as the observations and discussions made during the Seventh Review Meeting, including the actions implemented in the light of the Fukushima Daiichi accident.

With the aim of facilitating the understanding of this Report some information from previous reports is reproduced. Those aspects containing new information are addressed in italic.

1.1. GENERAL CONCEPTS

The Argentine Republic subscribed the Convention on Nuclear Safety, approved by a Diplomatic Conference in Vienna, Austria, on June 17th, 1994. In addition, on February 4th, 1997, the National Congress passed Law No. 24,776, approving the Convention adopted on September 20th, 1994. According to the provisions in Section 5 of the Convention, each Contracting Party must submit for its examination a National Nuclear Safety Report containing the measures adopted to comply with the corresponding obligations.

This *Eight* National Nuclear Safety Report describes the actions *that* the Argentine Republic has carried out since the Seventh Nuclear Safety Report was issued (May 2016) until *2019*, showing that it complies with the obligations derived from the Convention, in accordance with the provisions of its Article 4.

The analysis of the compliance with such obligations is based on the legislation in force, the applicable regulatory standards, the *conditions of the* licenses *issued* and other regulatory decisions. The corresponding information is described in the analysis of each of the Convention Articles constituting this Report.

Argentina has three NPPs in operation, CNA I, CNE and CNA II, which initiated their commercial operation in 1974, 1984 and 2016, respectively. Their corresponding net electric powers are 335 MW, 600 MW and 693 MW. These NPPs *supplied* about *4.7%* of the total electric power generated *in 2018.*

CNA I *(Central Nuclear Atucha I – Atucha I NPP)* is located about 100 km to the Northwest of Buenos Aires City *(Atucha site)*. The reactor is a PHWR type with a reactor pressure vessel (RPV). CNA I is fuelled now with slightly enriched uranium (0.85%). The reactor is moderated and cooled with heavy water.

CNA II *(Central Nuclear Atucha II – Atucha II NPP)* is located in the same Atucha site. Its reactor is also of the PHWR type with a RPV, loaded with natural uranium fuel and moderated and cooled with heavy water.

CNE *(Central Nuclear Embalse – Embalse NPP)* is located in the Province of Córdoba, approximately 110 km to the south of Córdoba City and 5 km to the Southwest of Embalse town. It is a CANDU type reactor, of pressure tubes type, loaded with natural uranium fuel, and moderated and cooled with heavy water. *CNE was out of service since December 31st, 2015 in order to implement its refurbishment and re-commissioning activities. Its first criticality was on January 4th, 2019, and since April 12th it is operating under the Commissioning Licence.*

The CAREM reactor prototype (CAREM) is an Argentine design and is under construction in the Atucha site, with a planned electric power of 25 MW. Its *main* design characteristics are an integrated primary system and passive safety systems.

Pre-licensing activities for the Fourth NPP have been initiated and a Memorandum of Understanding between the future applicant (Nucleoeléctrica Argentina Sociedad Anónima, NA-SA) and ARN was signed in November, 2018.

1.2. NATIONAL POLICY IN THE NUCLEAR FIELD

The national policy applicable to nuclear activities with peaceful uses is integrated by the provisions of the National Constitution and the legislation adopted by the National Congress by Law No. 24,804 enacted in 1997. The latter rules the Nuclear Activity along with Law No. 24,776 which approved the Convention on Nuclear Safety in 1997, and different laws related to the nuclear activity in accordance with treaties, conventions, agreements and international conventions.

In Argentina, the national nuclear policy was initially established by the Decree No. 10,936 enacted in 1950, that created the National Atomic Energy Commission (CNEA) with the objective of developing and handling nuclear technology. The control of the safety aspects of all nuclear activities performed in the country until the year 1994 were performed by the CNEA through its regulatory division, according to Law No. 14,467 and the Decree No. 842/58.

In 1994, the National Government assigned the exclusive performance of these duties to an independent state agency with federal competence.

This was implemented in the frame of the Decree No. 1,540/94 that reorganized the activities from the nuclear sector, and divided them into three entities; Nuclear National Regulatory Body (ENREN), Nucleoeléctrica Argentina Sociedad Anónima (NA-SA), and the National Atomic Energy Commission (CNEA), respectively responsible for the regulation, operation of facilities, and for research and development of the sector. Before that division, all these activities were developed by CNEA.

The abovementioned decree was then formally substituted by the federal Law No. 24,084 known as the "National Law of Nuclear Activity" sanctioned by the Argentine National Congress in 1997 and later complemented by the ruling Decree No. 1,390/1998. The Nuclear Regulatory Authority (ARN) was created, as the successor of the aforementioned ENREN.

Within this context, Law No. 24,804 is the current legal framework for the peaceful uses of nuclear energy in Argentina.

Article 1 of the mentioned Law, establishes that concerning nuclear matters the National Government, through the National Atomic Energy Commission (CNEA) and the Nuclear Regulatory Authority (ARN), shall define the state policy.

The nuclear policy shall meet all the obligations assumed by the Argentine Republic as a party to the Treaty for the Prohibition of Nuclear Weapons in Latin America and the Caribbean (Tlatelolco Treaty), the Treaty on Non-Proliferation of Nuclear Weapons (NPT), the Agreement for the Application of Safeguards involving the Argentine Republic, the Federative Republic of Brazil, the Brazilian-Argentine Agency for Accounting and Control of Nuclear Materials (ABACC) and the International Atomic Energy Agency (IAEA), in addition to the commitments assumed by Argentina as a member of the Nuclear Suppliers Group and the National Regime for the Control of Sensitive Exports (Decree No. 603/92).

Law No. 26,566, enacted in 2009 determined that NA-SA is the entity in charge for the life extension of CNE, the finalization of CNA II (both projects are already finished) and the life extension of CNA I which is an-going project. Also, said law mentioned that the construction, commissioning and operation of a fourth nuclear power plant, is under NA-SA responsibility.

1.3. NATIONAL PROGRAM CORRESPONDING TO NUCLEAR INSTALLATIONS

The construction and commissioning of the CAREM *prototype* for nuclear energy generation was declared of national interest *(Decree of the National Executive Power No. 1,107/2006)*, being the execution of the works necessary for the construction and commissioning under the technical responsibility of CNEA.

The National Government continues promoting nuclear activities in the country. Within this framework, the National Congress, through Law No. 26,566, 2009, declared of national interest the activities for the design, construction, licensing, procurement of goods and services, installation, commissioning, reception and put into commercial service of a Fourth NPP in Argentina, as well as all the necessary acts to enable the life extension of CNE, entrusting NA-SA the realization of these goals.

Law No. 26,566 also declares of national interest the design, implementation and commissioning of the CAREM prototype reactor to be built in Argentina, committing CNEA for that purpose.

In November 2018, a Memorandum of Understanding was signed between ARN and NA-SA, oriented to the construction of a fourth NPP. The fourth plant would be a HPR 1000 PWR unit (or Hualong I), with Fuqing unit 5 under construction in China taken as a reference design, and including design changes according relevant updates of Argentine and IAEA Safety Standards. China National Nuclear Corporation (CNNC) will be the supplier, while NA-SA will be the Responsible Entity, holding the Design Authority and the plant operator roles.

Activities related with the CNE´s Life Extension Project, to extend the plant life for another 25 years of full power operation as well as to increase the electrical power in about 35 MW, *were* carried out *and concluded. During the refurbishment outage* different design changes were introduced to improve safety, including post Fukushima requirements for severe accidents. The refurbishment shutdown started on December 31st, 2015 *and the start-up began in January 2019, reaching full power for the realization of commissioning tests by end of April 2019. Currently the plant is preparing the mandatory documentation for supporting the upcoming request of its Operating Licence.*

In addition, the Law No. 26,566 also contemplates the decision to proceed with the activities for the evaluation of the Long Term Operation of CNA I. Currently, NA-SA is working in the first phase of this project, i.e. the analysis and justification of the activities and modifications to be done for a safe continued operation during the LTO timeframe. (See section 3.6 for more details).

1.4. SUMMARY OF THE MAIN SUBJECTS CONTAINED IN THE REPORT

This *Eight* National Nuclear Safety Report has been performed in order to comply with Article 5 of the Convention on Nuclear Safety, and has been prepared, as much as possible, following the Guidelines Regarding National *Nuclear Safety* Reports Under the Convention on Nuclear Safety and the most significant conclusions introduced during the *Seventh* Review Meeting in *2017.* This means that the Report has been organized according to the Articles of the Convention and the contents as indicated in the above-mentioned Guidelines.

The information contained under the articles of the Convention, and its complementary Annexes, show the compliance of the Argentine Republic as a contracting party of this Convention, with the pursuant obligations assumed.

Chapter 2 of this report contains follow-up information on issues raised or requested by other countries at the *Seventh* Review Meeting. Chapter 3 includes detailed material that demonstrates how Argentina has implemented its obligations under Articles 6 to 19 of the Convention during the reporting period. For this purpose, the chapter enumeration is from 3.6 to 3.19 according to the corresponding articles of the convention. The Annexes at the end of the report contain expanded information from the main report.

The full text of the Argentine 1st, 2nd, 3rd, 4th, 5th, 6 th and *7th* Reports can be found on ARN's website and on the website of the International Atomic Energy Agency (IAEA).

Chapter 3 is divided in the following sections according to the Articles 6 to 19 of the Convention:

- Article 6 describes the actions adopted by the organization in charge of the operation of the NPPs (Licensee) in order to evaluate or improve safety. Such actions *are* a consequence of operational experience or in response to regulatory requirements.
- Article 7 presents the legal and regulatory structure that rules nuclear safety. It also describes the normative framework and the NPP licensing process as well as the regulatory control system.
- Article 8 shows functions, responsibilities, organizational structure, human and economic resources and personnel qualification of the Regulatory Body. It also explains the relationships with other interested organizations.
- Article 9 describes the Licensee's responsibilities and the controls required to verify the compliance with such responsibilities.
- Article 10 analyses the policies and the priority on nuclear safety established by the Regulatory Body as well as the Licensee.
- Article 11 deals with functions, responsibilities and the structure of the Licensee as well as the human and economic resources and the personnel qualification.
- Article 12 analyses the systems required to detect prevent and correct human errors.

- Article 13 shows the Licensee quality assurance program in the design, construction and operation of NPPs.
- Article 14 deals with deterministic and probabilistic safety assessments performed by the Licensee, and the safety assessments, evaluations and verifications performed by the Regulatory Body at every stage of the nuclear installation's lifetime.
- Article 15 describes the radiological safety criteria used, the existing rules on the subject, the authorised discharge limits, dose evaluations to workers of NPPs and to the public, and ALARA applications.
- Article 16 presents the laws, regulations and requirements existing in the country and their implementation in case of a radiological emergency at a NPP. It analyses the actions to be taken inside and outside each plant, by all the intervening organizations, with special emphasis in training exercises in the application of the emergency plan.
- Article 17 summarises the studies related to NPPs siting and site re-evaluation studies.
- Article 18 analyses the design and construction of NPPs and their compliance with the Argentine standards as well as application of good international practices, principles of defence in depth, diversity and redundancy. Besides, has been included the design improvements and the plant modifications which were already implemented or that are foreseen to be implemented to prevent BDBAs or to mitigate their radiological consequences.
- Article 19 deals with the mandatory documentation for NPPs operation, the technical support given to the installations, the operational limits and conditions, the maintenance activities and tests, the feedback mechanism of operational experience (OPEX), fire protection and relevant events communication, peer-review activities, and radioactive waste management.

Chapter 4 of this Report addresses the planned activities to improve safety.

Additional information is included in the following Annexes:

- **Annex I:** Conclusions about Argentina during the *Seventh* Review Meeting on the Convention on Nuclear Safety.
- **Annex II:** Answer to Questions or Comments - National Nuclear Safety Report - *2016*.
- **Annex III:** Main Technical Features of the Argentine Nuclear Power Plants in Operation.
- **Annex IV:** Principal Technical Characteristics of CAREM Reactor Prototype.
- **Annex V:** Examples of Lesson Learned and Corrective Actions Resulting from National and International Operating Experience and Events.
- **Annex VI:** Resume of NA-SA Quality Assurance General Manual Content.

The terminology contained in this Report is, in general, consistent with that used in the IAEA publications.

1.4.1. ACTIONS TAKEN IN THE LIGHT OF THE FUKUSHIMA DAIICHI ACCIDENT

This section summarizes the actions taken in the light of the Fukushima Daiichi accident, highlighting the Argentine´s continued efforts to strengthen the nuclear safety, in achieving the objectives of the IAEA Action Plan, the Convention on Nuclear Safety (CNS) and the Vienna Declaration, as well as to maximize the benefit of the lessons learned from the mentioned accident.

Additionally, member states having NPPs of the Ibero-American Forum of Radiological and Nuclear Regulatory Agencies (FORO), decided to conduct a stress test in each one of their NPPs similar to the one implemented by the Western European Nuclear Regulators Association (WENRA), with the objective to detect weaknesses in facing more extreme events and to propose the corresponding improvements.

The main stress test goal was to determine the NPPs safety margins, analysing their behaviour and response to extreme events that cause consequences beyond the design basis, such as long term loss of off-site and onsite power (station black out, SBO) and the loss of ultimate heat sink. The analysis also covers the capacity to manage such accidents.

In order to achieve this goal, a consensus was reached among the FORO member countries regarding the stress tests content and scope, so that each Regulatory Body required the mentioned stress test to the Licensees. It was also agreed a schedule for the preparation of the corresponding

National Reports (NR) and a cross peer review process among member countries leading to a FORO´s joint final report (FORO Report).

As was informed in the Sixth National Nuclear Safety Report the argentine Regulatory Body formalized this stress test by a regulatory requirement to the Licensee of CNA I, CNA II and CNE. This requirement consisted of a reassessment of the NPPs safety margins assuming the occurrence of a sequential loss of the lines of defence in depth caused by extreme initiating events and included:

- The design basis and licensing basis compliances review.
- The extreme initiating events conceivable at each NPP site.
- The loss of safety functions caused for each one of the extreme initiating events considered.
- Arrangement / disposal of structures, systems and components (SSCs) belonging to safety systems to assure they can continue fulfilling the corresponding safety function.
- The severe accident management program corresponding to each one of the extreme initiating events considered.
- 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 multiple reactors located in the same 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.
- Prevention, recovery and mitigation measures: automatic and operator actions for abnormal conditions; severe accident management and emergencies.
- Availability of the NPPs resources to face on-site and off-site emergencies on severe accident conditions. In particular, from the commencement of the event occurrence until the Regulatory Body takes charge of the emergency management, including the planning and action management considering the public protection and the corresponding communication.

In response to the regulatory requirement the CNA I, CNA II and CNE Licensee performed the above mentioned stress test and submitted to the Regulatory Body the corresponding Stress Test Reports. Later on, the Regulatory Body carried out an assessment of these Reports to verify compliance with the provisions of the regulatory requirement by considering, among others, the following aspects:

- Weaknesses identification.
- Improvement proposals to be implemented.
- NPP responses and effectiveness of preventive measures, highlighting any potential weaknesses and any cliff edge identified.
- Sequential loss of the existing defence in depth lines, independently of its probability of occurrence, assuming that the available measures were not effective to properly manage these scenarios.
- Analysis of the possibility of strengthening existing capacities to cope severe accident situations.
- Recovery and mitigation actions planned proposals.

The Argentinean Regulatory Body elaborated the National Report (NR), which contains the assessment results as well as the regulatory position regarding the proposal to implement the arisen improvements and modifications. The NR has been presented to the FORO to be reviewed jointly by all the members' countries. The review process allowed verifying that the assessment made by Argentina demonstrates the existence of suitable margins to fulfil the safety functions in case of the occurrence of the proposed extreme accidental situations.

Furthermore, as a FORO cross peer review process outcome, it was verified that the Argentinean NR meets the IAEA Action Plan and the FORO requirements. Later on a FORO Report was developed including the results of the peer review process which was approved by the FORO Plenary and, it was presented in the 2nd CNS Extraordinary Meeting in August 2012.

The improvements and modifications proposed by the NPPs licensees included an implementation schedule composed by short, medium and long term actions.

The general conclusion resulting from the stress tests performed by the Licensee and the Argentinean Regulatory Body is that there is a need for some regulatory actions but there are no relevant weaknesses that require urgent actions. It also concluded that the Licensee complies with both the design and licensing basis. For the purpose of increasing the capacity to respond to extreme situations the Licensee proposes to implement a set of improvements including the corresponding implementation schedules, which were considered acceptable by the Argentinean Regulatory Body. Therefore, it was issued a regulatory requirement requiring improvements and modifications referred to the issues above listed to be implemented in each NPP*. In the following paragraphs, a summary of the content of the mentioned requirement is presented.*

Most of the improvements corresponding to CNA I, CNA II and CNE have been accomplished or their implementation is an on-going activity. More details of the activities performed by the Licensee in relation with these requirements are presented in Sections 3.6, 3.14, 3.16 and 3.17.

1.4.1.1. EXTERNAL EVENTS *(See section 3.17.)*

- Seismic Margin Evaluation based on the corresponding updated seismic hazard (CNA I / CNA II).
- PSA based Seismic Margin Evaluation based on the corresponding updated seismic hazard (CNE).
- Installation of new seismic instrumentation (CNA I/ CNA II).
- Implementation of ECCS improvements arisen from seismic walkdown.
- Evaluation of safety margins in the case of flooding / low water level from earthquakes exceeding DBE.
- Implementation of easy fixes arisen from plant walkdown.
- New hydrologic and hydraulic study (CNA I / CNA II).
- Procedure for systematizing the outage of the plant in case of extreme low-water-levels (CNA I).
- Installation of an additional (fourth) pump to the river Water Cooling Ensured System (UK) (CNA I).
- Procedures for cooling the plant via the SGs in case of SBO and loss of the assured service water cooling system (CNA II).
- Re-evaluation of the consequences of probable earthquakes on the existing dam located downstream (CNE).
- Relocation of the site where is located the centre of the plant lighting system (CNE).
- Re-evaluation of the risk of tornadoes (CNA I / CNA II / CNE).

1.4.1.2. LOSS OF SAFETY FUNCTIONS (See sections 3.6. and 3.14.)

- New Emergency Power Supply System (EPS) (CNA I).
- Electrical interconnection between normal bars of CNA I and CNA II.
- External power supply protection devices (CNE).
- Upgrade of the EPS system (CNE).
- Replacement of the Class III DGs (CNE).
- Alternative power sources –MDGs- (CNA I / CNA II / CNE).
- Fuel elements integrity assessment (CNA I / CNA II / CNE).
- Extension of the UPS / batteries availability (CNA I / CNA II / CNE).
- Implement alternative water sources, and procedures associated:
  - o CNA I: process water refilling of the spent fuel storage pools by using an alternative reservoir (install an independent pump) and; replenishment of water inventory to the SGs through the secondary heat sink system.
  - o CNA II: alternative water reservoir that allows maintaining the water supply to heat removal through the SGs and cooling the spent fuel storage pool and; additional system to water replenishment of the spent fuel storage pools.
  - o CNE: installation of a device to connect a fire truck from outside the pool-building to replenish water; two mobile cisterns; water supply line to the calandria vault from outside the reactor

building; abnormal event procedure to provide additional water to replenish the dousing tank and SGs.

- Modifications to the emergency water supply (EWS) system (CNE).
- Improvement of ECCS reliability (CNE).
- Improvements of safety systems trip coverage / protection matrix (CNE).
- Install Instrumentation in the *secondary control room* (SCR), independent of main *control room* (MCR) to allow monitoring the level and temperature of water of the spent fuel storage pool. (CNE).

1.4.1.3. SEVERE ACCIDENT MANAGEMENT (See section 3.14.)

- Passive auto-catalytic recombiners (PARs) (CNA I / CNA II / CNE).
- Severe accident management program. Complete Severe Accident Management Guidelines (SAMG) (CNA I / CNA II/CNE).
- Strategies for reducing the pressure in the containment in case of severe accident using ventilation system (CNA I).
- Venting filtered containment system (CNE).
- Alternative cooling mode of the DGs by cooling towers (CNA I / CNA II).
- Review of procedures to increase DGs autonomy (CNA II).
- Install the instrumentation in the reactor, containment and spent fuel storage pools necessary for management of severe accidents (CNA I / CNA II / CNE).
- Implement a rupture disc on inspection port of calandria vault to be used to reduce the internal pressure in case of severe accident (CNE).
- *Cooling of the RPV external side for CNA I/CNA II (assessment of feasibility is an on-going activity).*
- *Containment filtered venting system CNA I/CNA II (effectiveness due to by-pass of containment is an ongoing activity).*

1.4.1.4. EMERGENCY PREPAREDNESS (See section 3.16.)

- Improve Emergency Plan considering lessons learned from Fukushima accident (CNA I / CNA II).
- *On-going assessment of the MCR and SCR habitability.*
- Source term verification and update (CNA I / CNA II / CNE).
- New fix and mobile satellite phones (CNA I / CNA II / CNE).
- Modification of the Procedure PS-101 “Conformation and function of the control of emergency internal centre - CICE” (CNA I).
- *Conditioning of a municipal emergency control centre away from the planning zone of the nuclear emergency (CNE).*
- Emergency exercises extended in time (long term) (CNA I / CNA II / CNE).
- Analysis of different facilities to retain the water used in decontamination tasks (CNA I, CNA II, CNE).
- Provision of portable electric generators with light columns and MDGs (CNE).
- Electric panel installed outside the service building with connection facilities to the emergency light equipment (CNE).
- Review and improvement of the procedures applied to the co-ordination and reception of supplies, equipment and additional personnel (CNE).
- Modification and improvement of the Internal Emergency Control Centre (CICE) building (CNE).
- Installation of an air recirculation filtering system in the SCR (CNE).

1.4.2. COMPLIANCE WITH THE PRINCIPLES OF THE VIENNA DECLARATION

On February 9th, 2015, the Contracting Parties meeting at the Diplomatic Conference of the Convention on Nuclear Safety adopted the Vienna Declaration on Nuclear Safety establishing principles for the implementation of the objectives of the Convention on Nuclear Safety to prevent accidents and to mitigate radiological consequences.

An Informal Technical Meeting of Nuclear Regulators convened by the Nuclear Regulatory Authority of Argentina was held in Buenos Aires on 16-17 November 2015 to exchange views on how to improve CNS reporting on the basis of the Vienna Declaration.

Argentine Republic has adopted the following principles of Vienna declaration:

1. New nuclear power plants are to be designed, sited, and constructed, consistent with the objective of preventing accidents in the commissioning and operation and, should an accident occur, mitigating possible releases of radionuclides causing long-term off site contamination and avoiding large early radioactive releases or radioactive releases large enough to require long-term protective measures and actions.
2. Comprehensive and systematic safety assessments are to be carried out periodically and regularly for existing installations throughout their lifetime in order to identify safety improvements that are oriented to meet the above objective. Reasonably practicable safety improvements are to be implemented in a timely manner.
3. National requirements and regulations for addressing this objective throughout the lifetime of nuclear power plants are to take into account the relevant IAEA Safety Standards and, as appropriate, other good practices as identified inter alia in the Review Meetings of the CNS.

In the following paragraphs a summary of the aspects addressing compliance with the principles of Vienna Declaration is presented. Also, the manner in which the information is included in this report is explained.

1.4.2.1. NEW DESIGN OF NUCLEAR POWER PLANTS

As was informed in previous National *Nuclear Safety* Reports, the Argentine regulatory authority is currently engaged in the licensing process of the CAREM 25 Prototype Reactor.

CAREM 25 design features have an enhanced implementation of the Defence in Depth (DiD) concept, and can therefore be considered to be an example of how the basic objective in the Vienna Declaration could be implemented in future projects.

A summary of basic design aspects of CAREM 25 Reactor in relation with DiD concept is presented below:

- Level 1 of DiD eliminates some initiating events with potential to threaten the reactor integrity. The integrated primary, featuring natural circulation and self-pressurizing, implies eliminating events as large LOCAs, LOFA and control rod ejection.
- Level 2 of DiD identifies the specific systems that prevent the demand of Safety Systems and in general that reduce the occurrence of fault sequences, namely risk reduction systems.
- Level 3 of DiD prevents initiating events from escalating to a severe accident, and it is unfolded in:
    - Sub-level 3A, with the goal of controlling *PIEs plus* single *failure* events within the Design Basis scenarios, accounts for both the short and the long term.
        - The Controlled State, namely grace period, is achieved by means of Safety Systems featuring passive driving forces (require no Power Supply) and is extended up-to 36 hours without requirement of operator intervention.
        - The second step, a Safe State is kept as long as necessary, by means of active systems actuated manually with no urgency, at any moment within the grace period.
    - Sub-level 3B, with the goal of controlling multiple failures or extremely rare events, accounts for two conditions in which the additional failures can take place.
        - For failures of the Safety Systems in Sub-level 3 A during step 1, the goal is Controlled State by means of diverse Safety Systems, also passive.
        - For failures in the Safe State (Sub-level 3A during step 2), the goal is to extend the grace period beyond 36 hours, by means of Safety Related Systems. It allows the operator intervention to recover the availability of the Safe State Systems.

- Level 4 of DiD mitigates conditions of core damage by the preservation of the confinement function, preventing releases to the environment. Design features dealing with preventing high pressure failure of the RPV, hydrogen deflagrations and detonations, corium-concrete interaction, and Containment failure in the long term (pressure increase is prevented by sprinklers and a Suppression Pool cooling system).

As it was mentioned in section 1.1. of this National Nuclear Safety Report, a Memorandum of Understanding (MOU) of a Fourth nuclear power plant (HPR 1000 - PWR reactor type) was already signed between NA-SA and ARN.

The main objective of the MOU was the establishment, since an early stage of the project, of the regulatory requirements and expectations in terms of licensing process and safety level that must be fulfilled by the design of the proposed plant and demonstrated through the Safety Analysis to be further submitted to ARN.

Regarding the design requirements, the MOU is in line with the Vienna Declaration as it states the mandatory fulfilment of AR standards, as well as the latest IAEA safety standards: Safety Fundamentals, General Safety Requirements (GSR) and Specific Safety Requirements specifically the Safety of Nuclear Power Plants: Design (SSR-2/1), Revision 1.

As part of defence in depth, analysis of Design Extension Conditions (DEC) shall be undertaken with the purpose of further improving the safety by:

- *enhancing the plant's capability to withstand events or conditions more challenging than those considered in the design basis, and*
- *minimizing radioactive releases harmful to the public and the environment as far as reasonably practicable, in such events or conditions.*

For project realization, ARN states in the MOU the need for a clear rationale connecting the engineering safety requirements for systems, structures and components, as derived from the Safety Analysis, with the safety classification following the IAEA, Classification of Structures, Systems and Components in Nuclear Power Plants (SSG-30).

The development of the MOU for Argentine next NPP and the CAREM project are practical examples that illustrate the strong commitment that Argentina has with the Vienna Declaration.

1.4.2.2. SAFETY REVIEWS FOR EXISTING NUCLEAR POWERS PLANTS

1.4.2.2.1. Periodic Safety Review

As it was informed in previous National *Nuclear Safety* Reports, since 2003 ARN has required a Periodic Safety Review (PSR) as a condition for renewal the *operating* license.

The requirement to develop PSR was included in the operating license of CNA I, CNA II and CNE. It was established that PSRs have to be developed every 10 years, with the scope described in the *IAEA SSG-25*, "Periodic Safety Review of Nuclear Power Plants". Approval of PSR results by ARN is a necessary condition for the renewal of *the operating* license *including for a long term operation period*. *PSR is used in Argentina for justification and development of the analysis of the minimum* modifications to be done for a safe continued operation.

CNA I presented its first PSR in 2014, being it the last one in the timeframe of design life. Although the plant had already implemented important design changes to upgrade the original design (CNA I was designed in the 60´s), the results of the review of safety factor "Design" identified new safety upgrade areas and improvements. Most of these activities and improvements will be implemented during a long scheduled outage as a condition for the Phase B of the long term period (see Section 3.6. and 3.14.).

Regarding the use of PSR as an integral tool for managing the cumulative effects of the ageing and development the Integrated Implementation Plan, the plant has enlarged the scope of the plant safety factors considering the regulatory expectation for the entire long term operation period. These include, but are not limited to the consideration of plant provisions to deal with DECs, as stated in the latest IAEA SSR2/1 Rev.1.

In the case of the CNE, due to the fact that the plant life extension programme *was developed by the designer*, the PSR is being developed as part of the safety assessments for that project. However, aging evaluations were completed and design improvements *were* introduced during refurbishment outage emerged from the results of safety evaluation made in other CANDU plants. As a condition for renewal of the operating license for the second cycle *in ten years time*, ARN requested a PSR with a comprehensive scope *as a baseline*. *See Section 3.14.*

1.4.2.2.2. Stress Test

In the frame of IAEA's action plan, and within the Iberoamerican Forum of Radiological and Nuclear Regulatory Agencies (FORO), comprehensive stress tests were carried out to determine the existing safety margins to cope with more extreme events, analyzing their behavior and the consequences for design extension conditions scenarios, such as station blackout and the loss of ultimate heat sink for a long term, as well as the capacity to manage such accidents.

The results of this analysis were presented in the Sixth *and Seventh* National *Nuclear Safety* Reports. The improvements to be implemented are summarized in Section 1.4.1. and are detailed in Section 3.6., 3.14., 3.16. and 3.17. of this National *Nuclear Safety* Report.

1.4.2.2.3. Operating Experience Feedback

During operation and commissioning stages, Argentinian nuclear power plants implement Operating Experience Feedback Programs taking into account internal and external events, as well as research findings.

Since 1998 it is required that the results of the implementation of these programs have to be submitted to the Regulatory Body, who controls that every necessary corrective or preventive actions are undertaken or planned.

The process of determining the applicability of the corrective or preventive actions resulting from an event in a given plant involves the utility in making an analysis of the design and the possible detection of weaknesses. An independent analysis of the most important events is performed by Regulatory Body staff.

More information about Operating Experience Feedback Programs is presented in Section 3.19., *as well as in Annex V.*

1.4.2.3. NATIONAL REQUIREMENTS AND STANDARDS

As it was informed in previous National *Nuclear Safety* Reports, ARN *is in an on-going* process of harmonization between the Argentinean Regulatory Standards and the IAEA Safety Standards. *Nevertheless*, Argentine Regulatory Standards are *already* consistent with IAEA's corresponding standards in general terms, taking into account that ARN has adopted a performance *or goal oriented approach.*

Moreover, Argentina participates actively in the IAEA standards committee´s activities and particularly in the international efforts to take account of the lessons learned from the Fukushima accident, in order to strengthen the nuclear safety in achieving the objectives of the IAEA Action Plan and the Nuclear Safety Convention, as well as to maximize the benefit of the mentioned lessons learned.

The Regulatory Body agreed with the Vienna Declaration on Nuclear Safety and adopted it in order to prevent accidents with radiological consequences and to mitigate such consequences should they occur. In this sense, ARN decided to carry out a normative framework *integral* review that includes addressing the Vienna Declaration in national standards.

The goals of the normative framework review are the following:

- Overall review of Argentina normative framework based on ARN regulatory experience as well as the international knowledge and Vienna Declaration. This review would include, if necessary, the modification of the existing standards and the development of new ones.
- Update the harmonization process of ARN regulatory standards in line with IAEA's standards, according to the Convention on Nuclear Safety and the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management.
- Facilitate the presentation and exchange of information on Argentine's standards, as part of preparation for the next Integrated Regulatory Review Service (IRRS) that will be carried out in Argentina.

The activities carried out by ARN to fulfil the above mentioned goals and the advances on normative framework review implementation during the period covered by this report are explained in Section 3.7.

CHAPTER 2

FOLLOW-UP FROM THE SEVENTH REVIEW MEETING

At the Seventh Review Meeting in 2017, the Rapporteur resumed the challenges and planned measures to improve safety for Argentina on specific topics to follow-up on this Eight Report. These topics are discussed in the following sections.

2.1. CHALLENGE 1: THE REGULATORY AUTHORITY TO PREPARE AND HOST THE IRRS MISSION IN 2018

In December 2014, the head of the ARN informed the IAEA the decision to initiate a process to receive an IRRS mission in the future, convinced that this step will be a relevant contribution to the demonstration, at national level, of the implementation of the objectives of the Safety Action Plan signed by all IAEA´s Member States.

In this context, during the reported period the following activities were agreed within the ARN:

- *Identification of a core group, called IRRS Group, to coordinate the process;*
- *Definition of working methods, working groups and responsibilities;*
- *Development of a diagnostic self-assessment process, to identify own strengths, weaknesses, opportunities and vulnerabilities; and,*
- *The IAEA´s SARIS methodology was adopted for the full self-assessment previous to an IRRS mission, and several of its phases have been undertaken (more info in section 4.6. of this report).*

The IRRS mission is scheduled for May 4th, 2020, as it can be seen at IAEA's webpage.

2.2. CHALLENGE 2: SALTO MISSION TO ATUCHA I

Argentina has selected for the methodological ordering of the Long Term Operation (LTO) project for Atucha I to follow the approach established in the applicable IAEA safety standards and guides. In particular the following IAEA safety standards were considered for both: the regulator, in order to define requirements and expectations, and the utility to prepare the documents to be submitted for a justification of a safe LTO: SSR 2/2 Rev.1, SSR 2/1 Rev 1, SSG-48 (and the former IAEA Draft Specific Safety Guide DS 485, Ageing Management and Development of a Programme for Long Term Operation of Nuclear Power Plants) and SRS 82 (Ageing Management for Nuclear Power Plants: International Generic Ageing Lessons Learned).

During the period 2016-2019, Atucha I received two PRE-SALTO missions, the last one took place during 23rd to 31st October 2018. Currently, SALTO mission is scheduled for 2020.

At the moment of the last Pre-SALTO mission, the regulatory requirements for the definition of LTO programme were established. These, included but were not limited to:

- *Implementation of improvements arising from the 2014 Periodic Safety Review (PSR),*
- *Comparison of Atucha I current design against the latest / modern German KTA design standards,*
- *Development of condition assessment of systems, structures and components (SSC) related to safety in accordance with the methodology defined by ARN,*
- *Completion of equipment qualification programme,*
- *Development of Time Limited Ageing Analysis (TLAAs) for structures and components belonging to systems safety classes 1, 2 and 3.*

The mission concluded that the plant made an important progress to implement a systematic ageing management review and prepare the plant for a safe LTO. Despite that some activities were partially implemented and some others were in planned stage, the team concluded that plant management is committed to improvement in the field of ageing management and plant preparedness for safe LTO and the appropriate implementation of all LTO related activities.

The team found some good performances, and identified fifteen (15) issues resulting in ten (10) recommendations or five (5) suggestions for improvement.

The most significant good performances are related to condition assessment reports in mechanical area, TLAA revalidation of the cumulative fatigue factor at locations between the containment and penetrations, and the implementation of a short term trending of preventive maintenance activities.

Regarding the recommendations, the most relevant are related to the methodology for scope setting for assessment of SSCs for LTO and implementation of a comprehensive equipment qualification programme. While the most relevant suggestion is related to the maintenance programmes for ensuring an effective management of ageing effects for LTO.

Additionally, at the moment of writing this National Report, SALTO mission was requested for November 3rd, 2020.

2.3. CHALLENGE 3: RESOLUTION OF ISSUES WITH ATUCHA I AND II RPV IN-VESSEL RETENTION AND EXTERNAL COOLING ARISING FROM FORO STRESS TESTS

The RPV external side cooling is considered as a means for retaining the corium inside the vessel in scenarios with extensive core damage. The strategy and its effectiveness are currently under analysis. In the past years, preliminary results obtained with RELAP5 / SCDAP were performed. These simple calculations were followed by more complex analysis with ANSYS / CFD code, performed for Atucha II NPP. The results of these analyses are not yet conclusive and therefore External Reactor Vessel Cooling can neither be totally ruled out nor considered a successful countermeasure.

CNA I

Given the complexity of a conclusive answer, a parallel course of analysis has been started, to assess possible stabilization of molten material inside sump, to avoid containment breach due to MCCI or at least delay it, so as to decrease consequences in public as far as reasonable achievable. This task is being performed jointly by Safety Analysis and Life Extension Project groups.

As the concept engineering of this strategy involves highly complex studies, it was proposed to incorporate it into the life extension programme.

CNA II

Nowadays, very detailed calculations with ANSYS / CFD code are being performed for SBO scenario. This analysis is intended to assess if a very Best Estimate calculation can lead to successfulness of the strategy. The calculation includes both cavity flooding phase and assessment of possible RPV failure due to thermal shock, and vessel cooling during relocation, to fully assess CHF occurrence considering RPV wall ablation mechanism and wall heatup influence, and also heat generation in the whole RPV wall.

In parallel, a set of scoping calculations with ANSYS / THERMAL code are planned, to get a complete view of the possible scenarios.

2.4. CHALLENGE 4: THE REGULATORY AUTHORITY TO CONDUCT LICENSING ACTIVITIES ON CAREM 25 SMALL MODULAR PROTOTYPE REACTOR UNDER CONSTRUCTION FOLLOWING PRINCIPLE 1 OF THE VDNS

The Argentine Regulatory Authority (ARN) is involved in the CAREM 25 Reactor licensing process, which is currently under construction. As reported in the previous National Safety Report, the design features of CAREM 25 have an improved implementation of the Defense in Depth (DiD) concept and, therefore, it can be considered as an example of how the basic objective of the Vienna Declaration could be implemented in future projects.

In reference to the development of the licensing activities ARN follows a proactive, rather than retrospective, approach accompanying the project realization. As in other licensing projects review & assessment, inspections and audits are performed following a safety oriented graded approach. In the *particular case of CAREM, the regulatory activities observed an enlargement in its scope for the purpose of analyzing the inclusion of the design aspects destined to comply the safety functions for events occurring in sub-level 3B of DiD. As it is mentioned in Section 1.4.2.1., the objective of sub-level 3B is the control of multiple failure events (design extension conditions), with a very low probability of occurrence, which defines a series of SSCs with particular engineering requirements designed to deal with these events.*

Special attention is given to the licensing of passive safety systems based on the knowledge of the physical phenomenon and the use of validated codes and standards for design and manufacturing.

The regulatory authority conducts licensing activities which include review & assessment, inspections, audits activities and enforcement actions, designed to verify compliance with safety requirements defined in the safety report. A so called "integrality concept" is used by which the connection between the engineering requirements for SSCs as derived from the Safety Analysis are verified to be consistent with those identified during the safety classification process.

2.5. CHALLENGE 5: EXTERNAL EMERGENCY CONTROL CENTRE LOCATED FAR FROM EMBALSE NPP

The design of the CNE External Emergency Control Centre, called Municipal Emergency Operational Centre (MEOC) was based on:

- *ARN requirements;*
- *WANO and IAEA recommendations;*
- *Experience gained in Benchmarking with an abroad NPP in France.*

The MEOC building design took into account the needs of information exchange and coordination of the operating groups belonging to the External Emergency Response Organization (EERO). It was conceived to have dedicated communication systems, specific equipments to predict the evolution of the accident, and its potential radiological consequences and to determine the protection measures to be applied in the population.

In case that the Internal Emergency Control Centre (IECC) buildings, next to the Embalse NPP, were inaccessible, the MEOC building will be allowed to be used as External Emergency Control Centre (EECC).

This MEOC building will be considered a safe area for the involved personnel. Therefore, it will support severe meteorological conditions and it will be constructed according to the Seismic Argentine Regulation "INPRES-CIRSOC 103 – Reglamento Argentino para Construcciones Sismo-Resistentes".

Until now, NA-SA is managing the efforts to find and acquire the suitable place to allocate the building according to all requirements to provide the most effective emergency response. In the meantime, it was proposed to subscribe an agreement between NA-SA and Almafuerte Firefighters Station located more than 15 km from the Embalse NPP. The temporary municipal emergency control will be equipped following the instrumentation and communication system to fulfil all the requirements up to the construction of the definitive facility is finalized.

2.6. REFURBISHMENT ACTIVITIES OF CNE

Among the main CNE refurbishment activities that were performed during the reported period, can be mentioned the following:

- *Reactor re-tubing: The removal and replacement of the fuel channels, including the corresponding pressure tubes, calandria tubes, end-fittings, feeders and others components like calandria tube inserts.*
- *Steam Generators replacement: Includes the replacement of the cartridges (SG´s tube plate and tube´s bundle), the pressure vessel envelope, and the steam drum internals (primary moisture separators) for the four SGs.*
- *Trip coverage´s improvement of the reactor´s shutdown systems (SDS # 1&2) for all original design basis accidents (DBA) and for some new events.*

- *ECCS´s reliability improvement: Includes the automatic initiation for smaller LOCA events, automatic switch from medium to low pressure injection, extra lines for injection, relocation of sump level sensors to improve measurement, etc.*
- *Reduce interface LOCA event frequency through the ECCS line.*
- *Automatic trips to protect main PHTS pumps.*
- *Improvement of seismic capacity of EPS and EWS systems.*
- *Improvement of the response to severe accident, by several design provisions: PARs to remove hydrogen from the containment and promote mixing within RB; installation of Filtered Containment Venting System to prevent the loss of containment structural integrity as a result of over-pressurization, installation of external pipe to add water inside the calandria vault and a rupture disc.*

Due to the safety significance of the tasks performed and the design changes implemented during the Refurbishment outage, a particular scheme of return to service was developed by ARN in order to take the systems and components to the normal operating condition.

For the systems that were not modified or intervened during the Refurbishment outage, they were returned to the service, taking them from the state or configuration in which they were during the outage to the normal operation state according to the current operational documentation.

For systems and components intervened or installed during the Refurbishment outage, 2 types of tests were defined:

- *Assembly Tests*
- *Commissioning Test*

Assembly tests were carried out after the work tasks (construction, assembly or reconditioning) to verify that, prior to the commissioning activities, the components, equipment, systems or parts of affected systems comply with the corresponding technical specifications.

Commissioning Tests were carried out to take components and systems of the CNE to the normal operating condition. In the case of the systems that were intervened due to design changes or new ones installed, tests had to be carried out to verify the correct operation of the system.

In this way, the commissioning of systems was divided into different phases:

Phase A: It contemplated all the activities carried out before initiating the removal of the state of guaranteed shutdown, as follow:

1. *Activities prior to the performance of the hydrostatic test of the Primary Heat Transport System (PHTS), from the completion of the construction phase.*
2. *Activities prior to the loading of fuel elements within the core, from the hydrostatic test of the PHTS.*
3. *Activities prior to the start of the first approach to criticality, from the beginning of fuel elements loading.*

Phase B: It covered from the beginning of the removal of guaranteed shutdown state up to 5% Full Power (PP).

Phase C: It consisted in increasing the power of the reactor until reaching 100% FP, executing all the tests defined for this interval.

ARN also established a series of milestones during the Commissioning process and defined prerequisites to be met to overcome them and continue with next commissioning stages. For example, for Fuel Elements loading, the Hydrostatic Test of the PHTS was required to be successful and communicate with enough anticipation to foresee the inspections of safeguards.

In the case of the authorization for the Guaranteed Shutdown Removal, criticality and increase up to 5% of full power, the Licensee was required to test the Containment structure. Also, the creation of an ad-hoc Committee composed of qualified persons with experience in the design, construction, start-up and operation of nuclear power reactors to evaluate the information of the Commissioning before its submission to ARN was required as well.

All the tests, either assembly or commissioning, were controlled by the ARN inspectors and the corresponding reports served as basis for the authorization of the different phases of the process.

Currently, the plant is operating under the Commissioning Licence and preparing the mandatory documentation to be submitted to ARN for support the request of the Operating Licence.

2.7. CNA I LIFE EXTENSION PROJECT

As it was mentioned in Seventh National Safety Report, Article 15 of Law No. 26,566 declared all the activities related to CNA I Life Extension as of national interest.

In order to license the LTO period of operation, ARN decided to split the project in two gradual phases:

- *Phase A, which objective is to maintain the current licensing basis as defined by the FSAR and PSR, both performed in 2014.*
- *Phase B, which objective is to increase, as far as practicable, the safety level of the plant.*

At the end of 2016, ARN defined the requisites and regulatory expectations to face a safe LTO – Phase A period.

The end of life (32 full power year), as defined by the designer (KWU), was reached in April 2018. Before reaching it, NA-SA submitted for further approval all the performed activities as requested by ARN:

- *Condition assessment of all in scope SSCs;*
- *Revalidation of Time Limited Ageing Analysis (TLAAs) related to the RPV structural integrity, SSCs for coping with the confinement function. Identification of TLAAs for structures and components to cope with the fundamental safety functions other than confinement;*
- *Development of equipment qualification master list (environmental, seismic and electromagnetic immunity) and the program for further qualification;*
- *Completion of safety factors from the last PSR (2014) according to the IAEA SSG-25;*
- *Plant implementation of relevant the recommendations which resulted from the condition assessments, in order to assure fitness for service of SSCs under the scope of Phase A;*
- *Implementation of new Fire – fighting's automatic systems.*

Additionally, in order to assess the preparedness of the plant for LTO, two pre-SALTO missions were conducted (see for more information Section 2.2.).

ARN reviewed and approved all the submissions and renewed the Operating Licence allowing the plant to enter in Phase A LTO period of operation.

Related to the Phase B, ARN issued a set of regulatory requisites and expectations on which NA-SA is currently working with the purpose to establish an integrated implementation plan that can increase the safety level of the plant.

For more information see Sections 3.6. and 3.14.

2.8. IMPLEMENTATION OF THE IMPROVEMENTS ARISEN FROM STRESS TESTS

Almost all of the improvements arisen from the stress test are already implemented. Summarized information can be found in Section 1.4.1. and detailed in Sections 3.6., 3.14., 3.16. and 3.17.

2.9. PROGRAMME OF CONSOLIDATION OF SAFETY CULTURE

As it was mentioned in the Seventh National Safety Report, the Licensee of CNA I, CNA II and CNE, has been developed a Programme of Consolidation of Safety Culture (PRACS in Spanish initials) to reinforce nuclear safety culture. The goal of the PRACS is to create a bridge between the concepts of Nuclear Safety Culture and actual performance in the installations. The programme defines management and implementation issues that require improvements by an eight-step strategy for each topic. The programme is based on work teams with team leaders with high credibility. The measurement of the PRACS performance is done through surveys, self-assessment and indicators.

In the period 2016-2018, the programme has broadened its scope covering more topics and involving the plants as well as the corporate organization. Currently, the programme covers topics such as operating experience, corrective actions follow up, self-assessment programme, fire protection, *emergency preparedness, human error prevention tools, risk management, ALARA, management indicators, equipment reliability, etc..*

PRACS Committees were set up at plant and corporate level, in order to follow up the programme. The Corporate PRACS Committee is chair by the Operational General Manager and it meets every four months to discuss the results of the programme and define corrective actions if are needed.

This programme was defined as an area of good performance during the Seventh Review Meeting of the Nuclear Safety Convention.

2.10. NORMATIVE FRAMEWORK REVIEW

As it was mentioned in the Seventh National Safety Report, ARN decided to address the Vienna declaration by incorporating it as a high level goal of a full-scope review in national standards, namely the normative framework review.

The activities carried out by ARN and the advances on normative framework review implementation during the period covered by this report are explained in Section 3.7.

2.11. PREPARATION OF IAEA IRRS MISSION

As was mentioned in Section 2.1. of this National Safety Report, the Regulatory Body has been undertaking the SARIS process previous to receive an IAEA IRRS (Integrated Regulatory Review Service) mission.

Since the National Report to the 7th Review Meeting the following activities in the preparation of an IRRS mission were carried out:

- *The National Workshop on IRRS mission and the IAEA Methodology and Tool for Self-Assessment of the Regulatory Infrastructure for Safety (SARIS) was held in Buenos Aires, from 25 to 27 April 2017, with the participation of more than 60 ARN´s staff, 7 members of other governmental offices and 3 IAEA´s experts.*
- *The SARIS methodology was implemented, completing the Respondent phase, being at the time of finishing with this National Report, in the phase of defining the Actions for the initial Action Plan for the mission (Ref: definition of phases according to the SARIS Methodology, IAEA´s SVS 37).*
- *Several ARNs staff members participated in some IAEA´s IRRSs missions, in various Training Courses for Reviewers in Integrated Regulatory Review Service missions, and also, ARN´s representatives took part of the Technical Meeting for the revision of the Integrated Regulatory Review Service Guidelines and the International Workshop on Lessons Learned from Integrated Regulatory Review Service (IRRS) Missions.*
- *The Preparatory Meeting of the IRRS mission was held in Buenos Aires, on 6-7 November 2018, with the participation of the mission leaders and IAEA´s coordinators. The preliminary results of the self-assessment were presented (8 detailed presentations).*
- *The ARN´s staff is devoting a continuous effort to the preparatory work and completion of the self-assessment for hosting an IRRS mission (see Section 2.1.).*
- *The IRRS mission was reprogrammed due to operative reasons (see Section 2.1.).*

Taking into account the Challenge 1 identified in the Rapporteurs Report of the 7th Review Meeting, it is worthwhile to mention that the self-assessment being conducted by ARN´s staff for the IRRS is considered by them a valuable tool contributing to analyse the organization and its practices in a systematic way, and of course, enhancing our views on how to explore actions to improve.

CHAPTER 3

COMPLIANCE WITH ARTICLES OF THE CONVENTION

Article 5 of the Convention requires that each Contracting Party shall submit for review a report on the measures it has taken to implement each of the obligations of the Convention. This report demonstrates the measures that Argentina has taken to implement its obligation under Articles 6 to 19 of the Convention. Obligations under other articles of the convention are implemented through administrative activities.

3.6. ARTICLE 6: EXISTING NUCLEAR POWER PLANTS

Each Contracting Party shall take the appropriate steps to ensure that the safety of nuclear installations existing at the time the Convention enters into force for that Contracting Party is reviewed as soon as possible. When necessary in the context of this Convention, the Contracting Party shall ensure that all reasonably practicable improvements are made as a matter of urgency to upgrade the safety of the nuclear installation. If such upgrading cannot be achieved, plans should be implemented to shut down the nuclear installation as soon as practically possible.

The timing of the shut-down may take into account the whole energy context and possible alternatives as well as the social, environmental and economic impact.

3.6.1. GENERAL

In Argentina, the NPP's Operating Licenses are granted for a limited period of time, typically no more than ten (10) years and for their subsequent renewal, a systematic safety reassessments based on IAEA SSG-25, Periodic Safety Review (PSR), has to be submitted to the regulatory body for approval of further continued operation.

PSR, as performed by the Licensee in Argentina, includes an assessment of plant design and operation against applicable current safety standards and operating practices, and has the objective of ensuring a high level of safety throughout the plant's operating lifetime including long term operation, as well. It is complementary to the routine and specific safety reviews conducted at nuclear power plants after significant events and implies an on step improvement of the plant's licensing basis. So, the current licensing bases for all NPPs were significantly enhanced since the original design.

Argentina has committed to fulfil the 2015 Vienna Declaration on Nuclear Safety (VDNS), which provides principles for implementing the Convention's objective: to prevent accidents and mitigate radiological consequences.

Principle (2) of the VDNS requires comprehensive and systematic safety assessments to be carried out periodically and regularly for existing installations throughout their lifetime in order to identify safety improvements that are oriented to meet the objective of the VDNS. Reasonably practicable or achievable safety improvements are to be implemented in a timely manner.

The above described illustrate that comprehensive and systematic assessments of the existing NPPs have been carried out and will continue to be carried out periodically in Argentina, resulting in numerous safety improvements that helped meet the objective in principle (2) of the VDNS.

3.6.2. EXISTING NUCLEAR POWER PLANTS IN ARGENTINA

Argentina has three NPPs under operating stage, CNA I, CNA II and CNE. Besides, the CAREM reactor prototype (CAREM), a low power NPP, is under construction.

CNA I, CNA II and CAREM share the same siting which is located approximately 100 km northwest from Buenos Aires city.

CNA I *is a PHWR reactor with* 335 MW (e) *power* and began its commercial operation in 1974. It is pressure vessel type moderated and cooled with heavy water. According to the original design CNA I was initially fuelled with natural uranium, but fuel elements of a new design having slightly enriched uranium (0.85% w 235U) were incorporated from 1995 to 1999, so that the reactor core is now fully loaded with slightly enriched fuel.

CNA II is a 745 MW (e) PHWR reactor, loaded with natural uranium and moderated and cooled using heavy water. It began its commercial operation in May 2016.

CNE, located in the Province of Córdoba, some 110 km to the south of the homonymous city, with 600 MW(e) began its commercial operation in 1984. It is a PHWR reactor of CANDU type, natural uranium loaded and heavy water moderated and cooled.

In Annex III, diagrams and some design characteristics of CNA I, CNA II and CNE are shown.

As was mentioned in Section 1.3. of this report, NA-SA is the Licensee in charge of the operation of CNA I, CNA II and CNE NPPs.

3.6.3. ACTIONS LEADING TO SAFETY IMPROVEMENTS

3.6.3.1. ROUTINE SAFETY REVIEW

As part of the regulatory system in Argentina, regulatory requirements and license conditions impose to the Licensee the mandatory responsibility for a continuous nuclear safety review. As stated in Principle 3 of Fundamentals Safety Principles, IAEA SF-1, safety is achieved and maintained by means of an effective Licensee's management system that ensures the fostering of a strong safety culture, regular assessment of performance and the application of lessons learned from experience. The knowledge gained from these activities becomes the driven force for some safety improvements.

Some of the routine activities in CNA I, CNA II and CNE that lead to safety improvements comprise, but are not limited to, the following:

- Documentation updating.
- Emergency Plan.
- Training and qualification of operating personnel.
- Quality Assurance Program.
- Ageing Management Program including plant programs.
- Severe Accident Management Program.
- Operating Experience Management Program.

While most of these activities are described along this National Report, some of them are explained below.

3.6.3.1.1. Documentation updating

Argentinean Regulatory Standard AR 3.9.1. –"General Criteria for Operational Safety in NPP" established that the SAR of nuclear installations must be updated each time that a plant design modification is performed, and once every five years.

In addition, the permanent documentation updating carried out in NPPs is based on the abnormal event evaluation performed, operating experience feedback, plant modelling with probabilistic techniques, identification of abnormal situations not specifically considered in the operation procedures, etc. This give rises to the implementation of new operational procedures or improvement to those already existing.

Also, mandatory documentation must be actualized according to the results of the Periodic Safety Review which is a requisite for renewal the operating license.

In this regard, CNA I *updated according to item 3 of IAEA SSG-25 the PSR performed in 2014 for a justification of a safe continued operation after April 2018. SAR and some of its topical reports like P-T curve and RPV structural integrity due to neutron embrittlement were updated. Besides, as part of this activity both deterministic and probabilistic safety analyses were updated, as well.*

In the case of CNA II, and due to the fact that is under normal operation since 2016 the documentation updating was done according to the operation experience gained. In this regard, some operational procedures and testing programme were modified.

In the case of CNE, *the FSAR was updated in order to reflect the design improvements and plant configuration as implemented from the scope of the life extension project. The update included, but was not limited to, chapter 2: Site Characteristics, chapter 6: Engineered Safety Features, chapter 15: Safety Analysis and Chapter 16: Operational limits and Conditions for Safe Operation. The structure and content of the FSAR was done following the IAEA GS-G-4.1 guidelines.*

Also, plant programs like: Maintenance, surveillance, inspection and testing in conjunction with plant procedures and guidelines, like Operating procedures and Procedures and guidelines for operating the plant during accidents, were also updated in accordance to the plant configuration that will face the second period of operation.

3.6.3.1.2. Ageing Management Program

Ageing Management Program is one of the mandatory documentation for granting a License and keeping valid the licensing basis in Argentina because deals with the physical effects of ageing of SSCs that can result in degradation of their performance characteristics with the potential consequence of decreasing the safety level of the plant.

Argentina follows the IAEA approach to set up the regulatory expectations for guiding the NPPs in defining effective ageing management programs which include the use of systematic approach to manage the effects of ageing in all stages: prevention, detection, monitoring and mitigation. Among ARN's expectation for effectiveness of ageing management program's development it can be stressed the use of the nine attributes as defined for example in the IAEA SSG-48.

In the case of CNA I, ARN used the opportunity of endorsing the long term operation to formally require the performance of a comprehensive ageing management review for all safety related SSCs in scope. CNA I performed a total of 47 condition assessment reports corresponding to 60 systems (SC1, SC2 and some SC 3 systems) for continued operation after the end of life as defined by the original designer. The results of these 47 reports were 374 recommendations categorized as non-critical for a safe long term operation. Most of the recommendations were improvements in plant programs like maintenance, in-service-inspection and water chemistry.

In addition, CNA I, CNA II and CNE are improving their current Ageing Management Program using the latest table of International Generic Ageing Lessons Learned (IGALL – AMPs) and checking the applicability of each one of the listed ageing management programs and verifying whether it exists already in the overall program or there is a need for a new one. As an example, it can be pointed out the need in CNA I to include AMP110 "PWR Boric Acid Corrosion" in order to address the recrystallization of boric acid problems, as it happened during the first cycle of operation.

In Argentina the current licensing basis for the existent NPPs has to be updated from the original (at least every 10 years) taken into account new regulations, codes and standards. This regulatory requirement gives an important role to the Obsolescence of SSCs Program as it refers that SSCs become out of date in comparison with current knowledge, standards and technology.

Both plants CNA I and CNA II have recently updated the Obsolescence Program using as a main reference IGALL, TOP401 "Technological obsolescence Program" – IAEA. The approval by ARN of these documents is an on-going activity. In the case of CNE the Obsolescence Program is based on TOP 401 as well.

3.6.3.1.3. Operating experience

Argentinean Regulatory Standard AR 3.9.1. –"General Criteria for Operational Safety in NPP" requires to the Licensee the performance of operating experience feedback, promoting the assessment of the events and proposing to the regulatory body the necessary modifications to the SSCs or procedures, in order to improve the safety of the installation.

Operating experience feedback is monitored and assessed by both, ARN and Licensee staffs, in order to enhance safety. Safety-significant operational events are evaluated for the purpose of identifying the immediate and underlying causes as well as defining and implementing corrective and preventive actions.

Operating Experience Programs were developed for the Argentinean NPPs in coherence with the latest IAEA SSG-50 guidelines where the operating organization encourages the plant personnel the reporting of all events, including low level events and near misses, potential problems related to equipment failures, shortcomings in human performance, procedural deficiencies or inconsistencies in documentation that are relevant to safety.

More information and examples of improvements implemented during the reported period can be found in Annex V.

3.6.3.2. SPECIFIC SAFETY REVIEW

NPPs in Argentina perform specific safety review as part of the renewal process (during the design lifetime and the long term operation period) of the operating license, as well as a reactive approach following a significant event. As it was already explained, the methodology for the specific safety review connected with the license renewal process is the PSR based on IAEA SSG-25 guidelines, while the methodology for the reactive approach following an event is graduated with the safety implications. This last may implies a vast set of activities, starting from operating experience and scaling up to peer review or the so called "stress test" performed after Fukushima.

PSR, as done in Argentina, is an integrated review that considers both operational performance trends and design safety, assessing the cumulative effects of plant ageing and modifications, as well as site-specific and organizational aspects. The scope is defined and agreed in between ARN and NA-SA covering all relevant safety issues and SSCs on the site encompassed by the operating license and their operation, together with the staff and the operator's organization. PSR in Argentina aims not only to confirm the safety level but also to improve it, in accordance to the new regulatory regulations and expectations.

3.6.4. IMPROVEMENT ACTIVITIES

In this section, some safety improvements that were implemented from *2016 to 2019* are shown.

3.6.4.1. CNA I IMPROVEMENTS

3.6.4.1.1. Construction of Dry Storage for spent fuel

In order to support the normal operation of CNA I during the period of LTO, a dry storage facility of Spent Fuel Elements (ASECQ) is being constructed as an extension of the spent fuel bay building.

The ASECQ will become a new building attached to and integrated into the existing building of the spent fuel bay building (hereinafter CP1). The building will include a silo located below the 0.50 m level, which will house the 2,754 spent fuel elements.

Through an opening excavated in the bottom wall of the CP1, the one built in the adjoining wall in the new building, and the extension of the existing bridge crane, it will be possible to move groups of spent fuel elements inside a properly shielded transport flask, to its new storage.

The benefit of the mentioned design arrangement is that the spent fuels do not leave Controlled Area and therefore the flasks for transferring them have fewer requirements: the structural safety must be verified by calculation, but not necessarily tested as required by AR regulation for transport in public roads.

The spent fuel elements can be stored wet or dry, and the process is reversible in the sense that they can go again to the wet storage waiting for future disposal. During the interim storage the spent fuel must maintain the same structure and integrity as those one that have never been deposited dry.

A sector of the silo will also serve to store irradiated reactor internal components or other elements (coolant channels, control bar guide tubes), which normally are located in the spent fuel bay of CNA I.

The facility will operate with the procedures that apply in the controlled area of the CNA I corresponding to CP1, and will also use the systems and services that are already in operation in it or adjacent to it, such as the normal power supply, the assured power supply system, the compressed air supply system and the sump system. The ventilation system will also be used through an interconnection with the current one.

The safeguard control of the spent fuels stored in the ASECQ will be added to the respective controls located in both bay buildings. The corresponding equipment will be set in an ad hoc building located in the building extension. Said location will have direct access from outside and not through the controlled area of the CNA I.

According to the regulation AR 10.1.1., Rev 3, the regulatory requirements applied in the licensing process of the ASECQ are similar than those one related to the installation modification process applicable during operation.

3.6.4.1.2. Emergency control room

In the frame of the LTO, CNA I installed a temporary emergency control room in the Secondary Heat Sink building. The design basis for this emergency control room is to allow the operators to fulfil the fundamental safety functions of shutting down the plant and keeping it in safe state under the event of inhabitability of the main control room due to toxic and / or corrosive gases, release of radioactive materials and sabotage.

The activities performed were:

- *Installation of computers with secured power supply in the room R 35-106.*
- *Definition of the signal list to be monitored to cope with the safety functions.*
- *Routing with physical separation two cables connecting the plant computer (Siemens 305) to the R 35-106.*
- *Installation of RESA button for fast shutdown system. Installation of a button to change the configuration of moderator system to residual heat removal function.*
- *Installation of communication systems: handies, mobiles, telephones, etc.*
- *Provision of the R 35-106 facility with radioprotection equipment and material.*
- *Development Instruction for operation and monitoring of the plant from the emergency control room.*

3.6.4.2. CNA II IMPROVEMENTS

3.6.4.2.1. Review and improvement of emergency procedures

Actions for optimizing fuel management for diesel generators, which aims to extend the operating time of emergency generators using the fuel of the auxiliary boiler, have already been implemented. In addition, a modification to the facility was carried out to supply fuel to weekly tanks of the diesel generators from the supply tank of the auxiliary boiler.

As these are tasks performed by areas belonging to the Emergency Response Organization (ERO), they changed from Severe Accident Management Guidelines to internal instructions of the ERO. The instructions have been developed.

With the objective of unifying severe accident guides of units I and II, it was decided that the Severe Accident Management Guideline SC 04-6 A for Severe Accident Management “Extension of Power Supply Time of Batteries”, belongs to the operations manual. This improvement has been implemented.

Instruction T-42 "Unavailability of the Main Control Room" will be incorporated into the Operations Manual, which will enable scram to be made from the Emergency Control Room located in the building of the Second Heat Sink, and bring the plant to a safe condition, until the Main Control Room can be recovered. This instruction is in process of approval.

3.6.4.2.2. Alternative water sources

The firefighting system of the construction site was identified as an alternative water reservoir. This system is currently connected to the plant firefighting system (SGA), which is the system that will be used to face a severe accident situation caused by the loss of heat sinks. Water replenishment of the spent fuel pools and water supply to the SGs in the long term will be possible through the plant firefighting system (SGA). *Some plant modifications were implemented* in order to allow the connection between the SGA system and the water supply *system of the SGs (GHC).*

Guidelines were drawn up within the framework of severe accident management:

- *"Water Injection to the Spent Fuel Pool*", which includes the replenishment of water through the demineralized water supply system (GHC) or through the firefighting water system (SGA).
- *"Feed and Bleed of Steam Generators", which includes the replenishment of water through the demineralized water supply system (GHC).*

3.6.4.2.3. Reposition of light water to the primary system from the volume control system

The objective is to have a volume control system (KBA) tank loaded with light water to replenish the inventories of the primary moderator system. Water is extracted from the storage tank by means of a circulation pump that injects into the common suction manifold of the high pressure charging pumps and, through one of them, it is sent to the main moderator system. In normal operation, the tank will be isolated. It has already been implemented and written.

3.6.4.3. CNE IMPROVEMENTS

Numerous improvements were implemented during the period *2016-2019* related to the CNE life extension project, as described in section 3.6.5.1.

3.6.4.3.1. Physical Protection System

For the Embalse Life Extension Project, the Regulatory Body has required improvements to the PPS, which are aimed to strengthen protection surveillance measures. The Surveillance is an essential aspect to the Physical Protection that has developed considerably during CNE Life Extension thus strengthening this area. This was reflected in:

- *Development and implementation of a secure information system and features reliable in data access.*
- *Ability to integrate information from different surveillance strategies.*
- *Improving the timing and quality of available data.*
- *Increased coverage of surveillance systems.*
- *Representativeness and integrity of information.*
- *Reinforcement of barriers and dissuasions.*
- *Upgrade of the access control system, through the increase of redundancy equipment.*
- *Incorporation of a contingency data center.*
- *Implementation of surveillance support systems.*
- *Storages for information protection.*
- *Equipment for communications (DLMR - Digital Land Mobile Radio) with the Response Force and training of Physical Protection personnel.*

A substantial improvement in video surveillance provides greater security against intrusion of internal and external agents.

3.6.4.3.2. Alternative water sources

The following alternatives water sources to face severe accident situations caused by the loss of heat sinks were implemented:

- Installation of a facility to connect a fire truck from outside the building pool to replenish water in the spent fuel storage was implemented in 2014 and the operators are trained on fire truck connection to this facility in the regularly basis.
- Two mobile cisterns containing 8,000 litres of stored water each were provided in 2015.

- Installation of a connection through a hose line from a fire truck to the ECCS pipes to allow water make up to the dousing tank for SGs' replenishment and reactor cooling beyond 7 days. This improvement was already implemented.
- Provide an additional fire truck with a capacity of 11,000 litres of water. This improvement was implemented in 2014.
- *Construction of a reservoir with assured 10,000 ton of water in the discharge channel in case of loss of the lake.*

3.6.4.3.3. Instrumentation & Control (I&C)

Installation of an I&C system in the secondary control room was completed in 2015 in order to allow monitoring the level and water temperature of the spent fuel storage pool, independent from the I&C system existent in the main control room, so that measurements could be repeated in both control rooms.

In addition, the spent fuel storage pool level and temperature measurements could be acquired outside the service and reactor building. These measurements do not use any electrical source.

Level measurement is obtained using a compressed air bubbler. In case of loss of compressed air, the system has an air tank with 48hs of autonomous operation. An external air source could be used to extend that period of operation in an event of loss of compressed air system.

Temperature measurement is obtained using a portable multimeter reading resistance from the RTD sensor installed in the spent fuel pool.

3.6.4.3.4. New systems providing redundancy during Refurbishment Outage

During CNE Refurbishment Outage *was* expected that normal redundancies for water supply and electric supply, particularly for spent fuel pools, *would* be impaired due to the execution of the planned works over those systems that provide such redundancies.

New systems *were* commissioned in CNE in order to provide sources of water in case of unavailability of Service Water System, and alternative electric energy in case of unavailability of the stand-by diesel generators.

3.6.5. NPPs LIFE EXTENSION

3.6.5.1. CNE LIFE EXTENSION

NA-SA, with the support of designer ex AECL, and currently CANDU Energy, performed a CNE life extension feasibility study in order to establish the scope, schedule and the necessary project investments.

Therefore, NA-SA developed a Plant Life Management (PLIM) and a Plant Life Extension (PLEX) program in order to maintain a high level of safety and plant performance during its life extension (extended period of 25-30 year). The PLEX program *was* divided into the following three phases:

Phase 1: Project Definition - assessment works. This phase consisted in carrying out plant condition assessments, which included aging and safety. These assessments allowed defining the activities to be undertaken in Phases 2 and 3. In general the aging assessment results indicated that the plant´s SSCs are in good conditions for life extension.

Phase 2: Project Engineering and material supply (preparation) – This phase consisted of engineering; purchasing management and preparatory activities prior to the plant refurbishment scheduled outage. This phase consisted in addressing and preparing the modifications and improvements arisen from the results of the assessments and analysis carried out in Phase 1 to start the detailed engineering of such design modifications (changes packages) approved by the Committee of Changes Approval as well as to prepare the corresponding works that was executed in Phase 3.

Phase 3: Project Implementation - field activities performed during the plant refurbishment outage. This phase consisted in the execution of the works (works packages) carried out in Phase 2 and included the corresponding refurbishment tasks like retubing and SGs replacement and the implementation of general modifications.

The Regulatory Body required to the Licensee, as a condition for licensing the CNE life extension to carry out a Periodic Safety Review (PSR), following the IAEA guidelines Safety Standards Series No. NS-G-2.10. Besides, an update of the mandatory documentation including the Final Safety Report was required.

The licensing process also followed the requirements established by the Argentine Regulatory Standards (AR standards), the applied Canadian Standards (Canadian Nuclear Safety Commission) due to Canada is the CANDU technology origin country, as well as international practices and, IAEA standards.

The scope and status of the tasks that were undertaken in the framework of the PLEX of CNE are detailed below:

Ageing Assessment

Plant ageing assessment, both Condition Assessment and Life Assessment, is part of an integrated strategy to assess the ageing degradation of the active and passive CNE components, which allow assessing the prognosis for life extension. Maintenance and inspection requirements can also be determined as well as the necessary upgrades or replacements needed to achieve the life extension in safe condition.

The ageing assessment indicated that CNE's SSCs are in good conditions for life extension. However, some obsolescence problems were identified which required implement several measures related to maintenance; inspections and testing in order to improve the ageing management program (see section 3.14.2.).

As a result of the ageing assessments conducted by NA-SA, there were generated 978 recommendations related to the repair or replacement of systems and components, and/or to new or modified plant programs and activities in order to address ageing and obsolescence issues pertaining to the systems and components. These recommendations were divided into several categories such as maintenance, inspections, obsolescence, design review, design changes, radiological protection and documentation.

Besides, the recommendations were divided into 3 categories according to their moment of implementation: before, during and after refurbishment outage.

ARN reviewed the management of the recommendations performed by NA-SA with special focus in those which are in the frame of time limited aging assessment.

Safety Assessment

The objective of the Safety Assessment is to determine issues susceptible to improvements, including among others, updating of the deterministic and probabilistic safety studies, the plant modifications incorporation and the use of updating computational models. The following are the safety assessments for life extension:

- Periodic Safety Review (PSR): specific requisites were elaborated by ARN according to the IAEA safety guide NS-G-2.10.
- Review and update of Deterministic Safety Analysis considering new accidental scenarios, updated models and acceptance criteria. Also, reactor repowering, new shutdown trips and others design changes.
- Probabilistic Safety Assessment: PSA L1 actualization was carried out with the objective of incorporating the new plant configuration after the refurbishment. A PSA L2 to evaluate the response of the containment and determine the radioactive release frequency was also developed. The results of PSA L2 were submitted and approved by ARN.

Review of CNE design against current national and international standards, regulations and practices

While the scope of CNE refurbishment was defined taking into account the comparison made for similar Canadian plants, ARN required NA-SA to perform a clause-by-clause CNE design review against 35 modern standards and practices, 13 of which correspond to Argentine normative and 22 to international regulations, mainly Canadian standards. NA-SA completed the review and submitted the results for approval to ARN between 2013 and 2015. In cases where discrepancies were found with regards to the current licensing and design basis, and with respect to modern standards and practices, ARN required NA-SA to provide an appropriate justification or to propose an improvement action to reduce or solve the gap.

Review and update of Hazard Assessments

In particular, a seismic re-evaluation/qualification of the CNE site is included. (See section 3.17.)

In addition, a Seismic Strategy for replacement or redesigning SSCs for CNE Refurbishment was elaborated by the Licensee with the support of the designer. This strategy was approved by the Regulatory Body and is based on the following:

- Classification of SSCs in three categories according to the extent of the modifications / changes in: Like-Like Replacement, Minor Modification, Major Modification or New Design.
- Assigning different seismic qualification requirements for each one of these three categories.

In any of the above cases and for the whole plant it was performed a PSA based Seismic Margin assessment (SMA) to determine potential vulnerabilities of the current design to face seismic event that could jeopardize the fundamental safety functions.

Safety Related Design Changes

Several improvements *were* implemented during CNE refurbishment. Those improvements are a result of review of Design Changes implemented in other CANDU plants, Probabilistic Safety Assessment, stress test post Fukushima, seismic re-evaluation, and review of CNE design against current standards.

The main safety related design changes *that were* implemented are:

- Improvements to Shutdown Systems (SDS) trip coverage; ARN required NA-SA that SDSs trip coverage will be improved so that it meets the following criterion: At least one trip parameter shall be incorporated into the sensing and control logic of each protective shutdown system for each of the serious process failures requiring shutdown action. The serious failures process considered are the listed in CNSC standard R-8 "Requirements for Shutdown Systems for CANDU Nuclear Power Plants".

  Implemented modifications are listed below:

  - Shutdown System #1
    - Addition of heat transport system low pressure trip.
    - Addition of moderator high level trip.
    - Addition of moderator low level trip.
    - Addition of steam generator low level trip on steam generators 2 and 3.
    - Regional Overpower Trip upgrade by addition of 21 new in-core flux detectors loops (Increase from 13 to 34 detectors).
  - Shutdown System #2
    - Addition of heat transport pump low speed trip.
    - Addition of moderator high level trip.
    - Addition of moderator low level trip.
    - Addition of steam generator low level trip on steam generators 1 and 4.
    - Addition of heat transport high pressure trip on reactor outlet headers 3 and 7.
    - Overpower Trip upgrade by addition of 16 new in-core flux detectors loops (Increase from 8 to 24 detectors).

- Improvements to Emergency Core Cooling (ECC) system:
  - Automatization of transfer from medium to low pressure stage.
  - Seismic qualification of the electrical supply for ECC pumps.
  - Triplication of dousing tank level measurement.
  - ECC initiation using sustained low primary heat transport system pressure.
  - Duplication of isolation valves of dousing tank.
  - Addition, modification and replace of other valves to improve reliability.
- Replacement of the current Emergency Power Supply System of 50 kW *by new 1600 kW / 2000 kVA Diesel generators, installed in a seismically qualified building*, which allows to provide energy seismically qualified to Emergency Water Supply System pumps and Emergency Core Cooling System pumps.
- Replacement of the two existing Emergency Water Supply diesel pumps by two higher flow, electric pumps, supplied from new Emergency Power Supply system. These pumps are able to feed the Emergency Core Cooling system heat exchanger.
- Addition of main heat transport pumps trip by sustained low pressure in the outlet headers, or high temperature in bearings, in order to protect the associated pipes of heat transport system.
- Addition of a line to add water in the calandria vault from outside of the reactor building.
- Addition of a rupture disk to the calandria vault to increase the release capacity, so that after a severe accident with loss of moderator heat sink, the pressure in the calandria vault does not increase beyond the design value.
- Design changes in the reactor fuel channels.
- Replacement of class III diesel power generators. The 4 existing diesel generators of 2,840 kW will be replaced by 4 new diesel generators of 5,200 kW, keeping the configuration 4 x 50%.
- Addition of autocatalytic hydrogen recombiners in the reactor building.
- Addition of a filtered containment venting system.
- Replacement of the Main Steam Safety Valves (MSSV) for new ones of higher capacity.

It is worth mentioning that all these improvements have been already implemented during the refurbishment outage.

Severe Accidents

There are two main aspects:

- Design improvements implementation to deal with severe accident evolutions and conditions. The improvements in the CNE design, related to severe accidents have two sources: the changes recommended by PSA L2 results, and those recommended from the operative experience in others CANDU plants.
- To develop and implement Severe Accident Management Guidelines (SAMG) specific for CNE, based on the Generic Guidelines developed by the CANDU Owner's Group (COG), with the designer support. These SAMG fulfil the Regulatory Body requirements related with the stress test which includes the consideration of the mitigation actions foreseen in order to prevent large radioactive releases as a consequence of damages to the reactor and the spent fuel pool. *The SAMG Program was completed by NA-SA.*

Environmental Qualifications (EQ)

A program in order to assure that safety related SSCs will be capable to perform their safety functions during normal and accidental conditions for the LTO was elaborated by the Licensee. For this purpose, it acquired the documentation used by the Hydro Quebec enterprise for the analysis and the implementation of the EQ Program in Gentilly II NPP, which has similar characteristics of design and construction, as well as a similar operation period to CNE. The corresponding procedures were adapted to the CNE by considering both the degree of applicability and the suitability of the plant existing documents.

During Refurbishment outage, ARN carried out inspections and verifications of the assembly and commissioning phases of systems with qualified components. Among the replaced components are:

- *Pressure transmitters and their auxiliary equipment corresponding to the Shut Down System No. 1 and No. 2 and the Emergency Core Cooling System.*
- *Ion chambers of Shut Down System No. 1 and No. 2 and the Reactor Regulating System. This included all corresponding cables, connectors and junction boxes.*
- *Approximately 2,000 meters of cables were replaced for environmental qualified cables. Most of them correspond to Instrumentation and Control; and the rest was distributed between medium and low voltage power cables.*
- *The internal elastomers were replaced by new ones of qualified EQ material in pneumatic valves of Shut Down System No. 2 and Emergency Core Cooling System. In addition, certain auxiliary equipment was replaced. For example, solenoid valves, pressure regulating valves and limit switches.*
- *RTDs related to the Emergency Core Cooling System were replaced for qualified components, including cables, connectors, junction boxes and accessories.*
- *Manual valve elastomers of the Emergency Core Cooling were replaced for qualified materials.*
- *The solenoid valves and cables of the Main Steam Safety Valves were replaced for qualified materials, including junction boxes and connectors.*
- *Cables and accessories of Emergency Core Cooling System pumps and its internal elastomers were replaced by qualified materials and components. The interplate joints of the heat exchanger were replaced by qualified equivalents as well.*

SGs replacement

This task included the replacement of the cartridges (lower assembly of the SGs), the tube bundles, the pressure vessel envelope, and the steam drum internals (primary moisture separators) for the four SGs. Only the steam drums including some internals remain from the original SG.

The new SGs were manufactured in Argentina under License and supervision of B&W by the local suppliers IMPSA, FAE and CONUAR, which were qualified under the plant designer supervision. New SGs included re-powering of the plant issues; the site seismic requalification and the design changes arising from the operating experience related to the PLIM program of the SGs.

The SGs replacement consisted in steam drum cutting and manipulating of old and new cartridges through the main equipment airlock. Once new SGs were positioned on their site, they were assembled. Old cartridges were stored in a special building for that purpose and supervised.

The connections of the PHTS pipes to the SGs nozzles were designed, installed, examined and inspected as required by the ASME code, section III, subsection NB.

The welding activities of the connections with the PHTS were carried out by the PCI company, belonging to the Westinghouse group, while the non-destructive tests were carried out by the company in charge of replacing the SGs.

The pressure tests of the welds were the pressure tests carried out on the entire PHTS which, as mentioned above, was carried out with a Hydrostatic Test of the PHTS at 120 kg/cm^2. With respect to the SGs, the results were satisfactory.

Digital Control Computers (DCCs) replacement

Although the Digital Control Computers used in CNE have had a good performance, replacement *was* required for LTO because of reasons related to obsolescence, maintenance personnel, ageing and expansion possibility. In order to deal with the replacement, CNE participated in a COG Joint Project for DCCs constructor qualification. The hardware and software supply *was* done by a qualified supplier and the nuclear designer, which also *supplied* the technical support for reception and commissioning tests of DCCs.

Replacement of the Moderator System Heat Exchangers (HXs)

The new HXs are equivalent to the *old* ones, mounted on the existent headers. The change in each of them *consisted* in replacing the tube bundle, the tube-plate, the housing and the shell. Regarding the tubes material, the alloy used contains a slightly higher percentage of chromium and nitrogen, which significantly improves its resistance to pitting. This design change *included* a tritium detection device, as a countermeasure for the occurrence of leakage from the moderator towards the process water.

3.6.5.2. CNA I LIFE EXTENSION

In December 2009, Law No. 26,566 was promulgated. In its Article No. 15, said law declares all the activities related to CNA I Life Extension as of national interest.

After the government decision was taken, NA-SA performed a plant feasibility study based on the fitness for a safe service of the long lived components, like the RPV. Comprehensive technical studies carried out on this component indicated that the safety margin that can be guaranteed on it makes technically feasible the Life Extension of the Plant.

Based on the foregoing, Resolution NA-SA No. 369/14 of August 12th, 2014 was issued, in which all the missions and functions for the Life Extension Project of the CNA I NPP were enunciated. In turn, the decision of NA-SA to carry out the Extension of Life (PLEX) of the CNA I was communicated to the Regulatory Body.

Taking into account the lifetime limit due to the RPV's fluence, and adopting a conservative approach, a life extension period of up to 56.25 full power years (FPY) was proposed. This was communicated as a proposal to ARN.

NA-SA has selected the approach established by the applicable IAEA safety standards: SSR-2/2 (Rev.1), SSR-2/1(Rev.1) and SSG-48 for the methodological ordering of CNA I´s PEV.

The aspects mentioned in the Safety Reports Series No. 82, "Aging Management for Nuclear Power Plants: International Generic Aging Lessons Learned (IGALL)" are also considered. NA-SA receives advice and methodological support through the IAEA's SALTO (Safety Aspects of Long Term Operation) Program.

Based on the above mentioned decision, ARN defined two phases within the long term operation (LTO) ("A" and "B") with different regulatory expectations for a safe continued operation. The regulatory requirements for phase "A" are oriented to maintain the licensing basis according to the latest 2014's PSR and for phase "B" the requirements are focused in upgrading, as far as practicable, the safety level of the plant.

The design end of life as defined by the designer was reached in April 2018 and the current operation of the plant is under the phase "A" of LTO. The corresponding authorization was issued based upon the activities performed by NA-SA to meet the regulatory requirements for phase "A":

- *The identification of safety-relevant structures, systems and components (SSC) was completed, as well as the current condition assessments of these SSC according to the defined scope, methodology and criteria. Also, an independent review was requested by ARN and completed by NA-SA.*
- *The Safety Analyses related to the integrity of the RPV, as well as the TLAAs of structures and components that fulfil the containment function were performed. The TLAAs of structures and components related to the integrity function of the pressure boundary of the primary coolant, as well as the function of shutting down and maintaining safe shutdown condition of the reactor were identified.*
- *The equipment environmental qualification manual, the environmental qualification programme and the master list of equipment and components that require environmental qualification were carried out in accordance with the established methodology.*
- *The activities related to the safety factors of the last PSR were completed.*
- *During the scheduled shutdown in 2018, the tasks of regulatory interest were completed, among which the recommendations arising from the condition assessment stand out, as well as the implementation of new automatic fire extinguishing systems.*

After the requirements were met, ARN issued an amendment to the plant's Operation License, establishing a validity of five equivalent full power years or ten calendar years since the approval of the last Periodic Safety Review on September 29th 2014, whichever occurs first.

In preparation for justification a safe phase "B", engineering studies are being developed and design upgrades are analysed based on the PSR's Global Assessment methodology, as stated in IAEA SSG-25. It is expected that this Global Assessment and the derived Conceptual Implementation Plan be finished and submitted to ARN by March 2020.

This Conceptual Improvement Plan will include all those projects that allow improving safety level as much as possible/practicable outlined in the modern regulations and the state of the art. Said plan will also contain the prioritization of said tasks based on the aforementioned methodology.

In accordance with the IAEA guidelines adopted for the Life Extension and the Policy and Program document for the Long-term Operation of the Atucha I NPP, in addition to the activities of the Conceptual Improvement Plan, the necessary tasks and studies are being developed to be able to guarantee in the proposed extension period, the proper management of aging of the set of systems and components (SC) that fulfill safety functions in the plant, or of those whose failure could prevent compliance with the safety functions (Scope of LTO).

The regulatory decision to proceed with the Long Term Operation of CNA I will be based on the acceptance of the Conceptual Implementation Plan at the extent it can improve the robustness of the current Defence in Depth Concept.

The main activities that are being carried out in prospect of a Long Term Operation – stage "B" include but are not limited to:

- *Modernization of the Reactor Protection System.*
- *Consideration of consequential failure – Implementation of Protective goals.*
- *Improvement of habitability of Main Control Room and implementation of a Secondary Emergency Control Room.*
- *Demonstration of Leak Before Break behaviour.*
- *Physical separation of components belonging to safety related systems.*
- *I&C renewal process.*
- *Aging Management Review of structures and components (SC) that are within the scope of LTO, including the determination and implementation of Ageing Management Programs needed to have a credited ageing management during the long term operation.*
- *Identification and revalidation aging studies and calculations for SC whose degradation of functions can be calculated as a function of time (Time Limited Aging Analysis).*
- *Building Improvements of the Second Heat Sink – SSC.*
- *New suction filters for safety injection pumps – TJ.*
- *Dry storage of used fuel elements.*
- *Improvements in fire protection.*
- *Establishment of equipment qualification in CNA I.*

Some others activities could be added depending on the results of the updated PSR, taking into account ageing assessment, operating experience, comparison with international standards, etc.

Meanwhile, NA-SA and the Regulatory Body will sign a Licensing Basis Document to define the licensing *and activities* of the project.

3.6.6. OPINION CONCERNING THE OPERATION CONTINUITY OF NUCLEAR INSTALLATIONS

During this reporting period, NPPs were operating with acceptable safety margins, complying with the Regulatory Standards related to design and operation. The *concept* of defence in depth of the *existing* NPPs remains acceptable, *like the one in CNA II or was/is being upgraded like in CNE and CNA I*. *Besides,* the ARN's requirements were fulfilled. NA-SA and the ARN, each within their corresponding roles and responsibilities, ensure that the NPPs are operating under the conditions and within the safety margins included in the mandatory documentation.

3.6.7. COMPLIANCE WITH THE OBLIGATIONS IMPOSED BY THE CONVENTION

From the preceding considerations, it becomes evident that since the beginning of nuclear activities in Argentina, continuous and detailed safety assessments and improvements are carried out in NPPs.

Furthermore, NPPs were operating with acceptable safety margins, complying with the Regulatory Standards related to design and operation, and the level of defence-in-depth of the NPPs remains acceptable. Therefore, the country complies with the obligations imposed in Article 6 of the Convention on Nuclear Safety.

3.7. ARTICLE 7: LEGISLATIVE AND REGULATORY FRAMEWORK

1. *Each Contracting Party shall establish and maintain a legislative and regulatory framework to govern the safety of nuclear installations.*
2. *The legislative and regulatory framework shall provide for:*
   - i. *the establishment of applicable national safety requirements and regulations;*
   - ii. *a system of licensing with regard to nuclear installations and the prohibition of the operation of a nuclear installation without a license;*
   - iii. *a system of regulatory inspection and assessment of nuclear installations to ascertain compliance with applicable regulations and the terms of licenses; the enforcement of applicable regulations and of the terms of licenses, including suspension, modification or revocation.*

3.7.1. NATIONAL LEGISLATIVE FRAMEWORK

A national legislative framework has ruled the nuclear activity in Argentina since the decade of 1950, as it was mentioned in the previous national reports *and in section 1.2.*

The National Commission of Atomic Energy (CNEA) was created in 1950 by Decree No. 10,936/50. One of CNEA's specific responsibilities was the control of all public and private nuclear activities to be performed in the national territory.

Later, various decrees defined CNEA's competence also as the Regulatory Body for nuclear and radiological safety, particularly concerning the individual and environmental protection against the harmful effects of ionizing radiation, the safety of nuclear installations and the control of the use of nuclear material. In this regard, the specific regulations were Decree-Law No. 22,498/56, ratified by Act No. 14,467 and Decree No. 842/58.

Law No. 14,467 determined CNEA's competence to issue the necessary regulations for the permanent control of the activities related to radioactive substances, and to provide the necessary means to control the existence, marketing and use of materials related to peaceful applications of atomic energy.

Decree No. 842/58 approved and put into force the "Regulation for Using the Radioisotopes and Ionizing Radiation" and made it effective to rule the use of radioactive materials and radiations they emitted or which were originated by nuclear reactions and transmutations. The use of X-rays generators was excluded from the competence of the CNEA and entrusted exclusively to the Ministry of Health.

The sustained growth of nuclear activity in the country made it necessary to strengthen the independence of the Regulatory Body with respect to the other activities of the CNEA.

In 1994 the Government decided that the regulatory function and control of nuclear activities would be preserved at state level, and formally independent from promoters and users.

Based on these considerations, the National Executive Power, supported by Law No. 23,696/89 and by Section 99 Paragraph 1 of the Constitution, created the National Board of Nuclear Regulation (ENREN) by Decree No. 1,540/94 to perform all the regulatory functions of the nuclear activity (formerly within the competence of CNEA's regulatory branch), transferring to ENREN the corresponding staff, equipment and facilities. As from 1997, the ENREN adopted the present denomination of Nuclear Regulatory Authority (ARN).

The nuclear activities of the Argentine Republic are developed within a legal framework with one main rule: the National Law of Nuclear Activity No. 24,804 enacted in April 1997, with its Regulatory Decree No. 1,390/98.

The legal framework is formed by the National Constitution, the treaties and conventions, laws, decrees and resolutions as stated below and by the Regulatory Standards (see section 3.7.2.1.). The Argentine Republic has adhered as contracting party to a number of bilateral and multilateral international instruments, which imply different commitments and obligations for the State in the nuclear field *with peaceful purposes.*

The present most relevant National Legislative framework put in force for peaceful uses of nuclear energy in the country is listed in chronological order.

- *Law No. 17,048: Nuclear Damage. Vienna Convention on Civil Liability for Nuclear Damage, 1966.*
- *Law No. 24,776: Convention on Nuclear Safety, 1997.*
- Law No. 24,804, 1997, “National Nuclear Law”. *Decree No. 1,390/98.*
- Law No. 25,018, 1998, *Radioactive Waste Management System.*
- *Law No. 25,313, 2000: Protocol to Amend the Vienna Convention on Civil Liability for Nuclear Damages and the Convention on Supplementary Compensation for Nuclear Damages.*
- Decree No. 981/05, Licensee NA-SA, to conform the Atucha II Unit of Management, for completing the construction and putting into operation CNA II.
- Decree No. 1,107/06, declares of national interest the construction and operation of the CAREM NPP.
- Resolution ARN No. 107/2007: Limits the collective doses per unit of practice for nuclear power plants applicable to the discharge of radioactive effluents from NPPs.
- Law No. 26,566, 2009, declares of national interest the activities to construct a fourth NPP, the life extension of Embalse NPP and the CAREM Reactor Prototype.
- Resolution ARN No. 352/13: Authorization for Use of the Site and Construction of CAREM Reactor Prototype.
- Resolution ARN No. 238/14 issued the Commissioning License of CNA II. *Later modified by Resolution ARN No. 684/15.*
- Resolution ARN No. 302/16 issued the Operating License of Nuclear Power Plant Presidente Dr. Néstor Carlos Kirchner (CNA II).
- *Resolution ARN No. 477/16 issued an addendum to the Operating License of CNE authorizing to the Licensee to proceed with the system interventions in accordance to the Life Extension project scope.*
- *Resolution ARN No.157/2018 issued an addendum to the Operating License authorizing the Phase A of the Long Term Operation.*
- *Resolution ARN No. 01/2019 issued an addendum to the Operating License of CNE authorizing the removal of the safe shutdown state and to proceed with the commissioning activities.*

3.7.2. NORMATIVE FRAMEWORK

3.7.2.1. INTRODUCTION

Law No. 24,804/97 empowers the Regulatory Body to issue and establish the standards, which regulate and control nuclear activities, of compulsory application, along the whole national territory.

The first Regulatory Standards related to nuclear power plant licensing were initially produced more than thirty years ago and were known as "CALIN Standards" (acronym for Consejo Asesor para el Licenciamiento de Instalaciones Nucleares - Advisor Council for the Licensing of Nuclear Installations, a body belonging to the regulatory branch of CNEA in that period).

Over time, a normative system was established comprising subjects such as radiological and nuclear safety, safeguards of nuclear materials and physical protection. The system, known as "AR Standards" (AR Standards for Regulatory Body), has at present 65 regulatory standards of which 31 are related to NPPs. The codes and names of the before mentioned 31 regulatory standards are shown in Table 3.7.1.

Additionally, there is a permanent Regulatory Body activity that reviews and updates the regulatory standards.

Table 3.7.1. - AR Standards concerning nuclear power plant licensing

AR Code	Name
0.0.1.	Licensing of Type I Installations
0.11.1.	Licensing of personnel of Type I Installations
0.11.2.	Psychophysical aptitude requirements for Specific Authorizations
0.11.3.	Retraining of personnel of Type I Installations
3.1.1.	Occupational exposure in nuclear power plants
3.1.2.	Limitation of radioactive effluents in nuclear power plants
3.1.3.	Radiological criteria relating to accidents in nuclear power plants
3.2.1.	General safety criteria in the design of nuclear power plants
3.2.3.	Nuclear power plant fire protection
3.3.1.	Nuclear power plant reactor core design
3.3.2.	Nuclear power plant heat removal systems
3.3.3.	Nuclear power plant primary pressure circuit
3.3.4.	Nuclear power plant fuel performance
3.4.1.	Safety-related protection and instrumentation system in nuclear power plants
3.4.2.	Nuclear power plant shutdown systems
3.4.3.	Nuclear power plant confinement systems
3.5.1.	Emergency electric power supply in nuclear power plants
3.6.1.	Nuclear power plant quality system
3.7.1.	Documentation to be submitted to the Regulatory Authority prior to the commissioning of a nuclear power plant
3.8.1.	Pre-nuclear commissioning of nuclear power plants
3.8.2.	Nuclear commissioning of nuclear power plants
3.9.1.	General criteria for operational safety in nuclear power plants
3.9.2.	Communication of significant events in nuclear power plants
3.10.1.	Protection against earthquakes in nuclear power plants
3.17.1.	Nuclear Power Plant decommissioning
10.1.1.	Basic Radiation Safety Standard
10.10.1.	Site Evaluation for Nuclear Power Plants
10.12.1.	Radioactive Waste Management
10.13.1.	Basic standard for the physical protection of nuclear materials and installations
10.14.1.	Assurance of non-diversion of nuclear materials and of materials, installations and equipment of nuclear interest
10.16.1.	Transport of radioactive materials

Moreover, the normative system has at present 10 regulatory guides; 6 of them related to NPPs and listed in Table 3.7.2.

Table 3.7.2. - AR Regulatory Guides concerning nuclear power plants

AR Code	Name
AR 1	Dosimetric factors for external irradiation and internal contamination and emergency levels in food
AR 3	*Specific functional* conditions to be verified by the specialized physician according to psychophysics performance score
AR 8	Generic clearance levels
AR 10	Specialized training programme and specific training for licensing of personnel of Type I radioactive installations
AR 13	Storage of radioactive waste
AR 14	*Development and Design of an Radiological Environmental Monitoring Plan*

During the period covered by this report, the standard AR 10.12.1."Radioactive Waste Management" was revised to update concepts and requirements included in international regulations and, as far as relevant, the recommendations of the IAEA were taken into account.

Also, the Regulatory Guide AR 14 "Development and Design of a Radiological Environmental Monitoring Plan" to be applied to type I installations was issued to facilitate the Regulatory Standard AR 10.1.1 fulfilment.

3.7.2.2. COMPLIANCE WITH THE PRINCIPLES OF VIENNA DECLARATION ON NUCLEAR SAFETY

According to the previous Nuclear Safety Reports, ARN performed a process of harmonization between the Argentinean Regulatory Standards and the IAEA Safety Standards. As result of such harmonization it was concluded that Argentine Regulatory Standards are consistent with IAEA's corresponding standards, taking into account that ARN has mainly adopted a performance criterion.

Moreover, Argentina participates actively in the IAEA standards committees and particularly in the international efforts to take account of the lessons learned from the Fukushima accident. This is reflected in our normative reviews that keep track of the updated adjustments of IAEA standards when assessing the Argentinean normative, in order to strengthen the nuclear safety in achieving the objectives of the IAEA Action Plan and the Nuclear Safety Convention.

The Regulatory Body agreed with the Vienna Declaration on Nuclear Safety in the understanding that it is the permanent goal of Nuclear Safety to prevent accidents with the radiological consequences and to mitigate such consequences should they occur. In this sense, ARN decided to address the Vienna Declaration by incorporating it as a high level goal of a full-scope review in national standards, namely the normative framework review.

The goals of the normative framework review are the following:

- Overall review of Argentina normative framework is based on ARN regulatory experience as well as the international knowledge and Vienna Declaration. This review would include, if necessary, the modification of the existing standards and the development of new ones.
- Update the harmonization process of ARN regulatory standards in line with IAEA's standards, according to the Convention on Nuclear Safety and the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management.
- Facilitate the presentation and exchange of information on Argentine's standards, as part of preparation for the next Integrated Regulatory Review Service (IRRS) that will be carried out in Argentina.

The activities carried out during the period of this report in order to fulfil the above mentioned goals are the following.

First phase of Regulatory Framework Review included:

- *Review of the existing legal and regulatory framework. The main objective of this activity was a comparison in between the requisites stated in IAEA, Safety Standard GSR Part 1 (Rev. 1) Governmental, Legal and Regulatory Framework for Safety and the nuclear framework in Argentine Republic.*
- *Assessment of the regulatory framework carried out by the relevant regulatory areas of ARN. Licensing and control areas of ARN performed an exam of applicable standards and guides taking into account regulatory experience and operative experience, requisites of the Conventions signed by Argentine Republic and IAEA recommendations.*
- *Identification of gaps in the existing corpus of Standards and Guides.*
- *Findings report and diagnosis of the situation.*
- *Development of an action plan.*

In relation with the normative framework applicable to nuclear power plants, the action plan to be completed in the 2017-2022 period include the elaboration of 18 new standards and 7 new guides; and the revision of 11 standards and 1 guide.

At the beginning of 2019 the projects corresponding to the elaboration or revision of 10 standards and 1 guide applicable to nuclear power plants have been initiated and are in different stages of the process.

The following are some examples of standards that are being developed and the corresponding approaches adopted for their preparation:

- *Glossary that incorporates new terms and harmonize terms used in different standards.*
- *Management system for installations and practices, that is a basic standard based on IAEA Safety Standard No. GSR Part 2.*
- *Preparedness and Response for a Nuclear or Radiological Emergency, that takes into account recommendations of IAEA Safety Standard No. GSR Part 7.*
- *Licensing of Class I installations, that is a revision of the existing regulations in order to incorporate the regulatory experience on the nuclear reactors licensing process in the last years.*
- *Safety requirements for nuclear reactors construction, it is a new standard that contains requirements based on the regulatory experience obtained during the construction of CNA II and CAREM, as well as, important design changes of CNE NPP; taking into account IAEA Safety Standard No SSG-38.*
- *Format and Content of Safety Analysis Report for NPP, that incorporates the requirements already issued to the licensees to complete this kind of documents.*
- *Periodic Safety Review for NPP that contains the requirements issued for PSR of CNE and CNA I.*
- *Safety requirements for the design of NPP, that is a new standard that contains the revised requisites of the existing AR standards for NPP design and takes into account requirements of IAEA Safety Standard No. SSR-2/1 (Rev. 1).*

3.7.2.3. BASIC CONCEPTS

The Argentine Regulatory Standards are based on a set of fundamental concepts, which are part of the performance approach philosophy, sustained by the regulatory system, concerning radiological and nuclear safety, safeguards and physical protection.

Such fundamental concepts, related to radiological and nuclear safety, are developed in the following sections.

3.7.2.3.1. Deterministic and probabilistic aspects of the Regulatory Standards

The installation's radiological and nuclear safety is conceivable by means of two approaches: a deterministic and a probabilistic one. Both approaches are complementary and are being used in radiological and nuclear safety in a balanced manner. In this sense the ARN adopted, more than three decades ago, a probabilistic criterion for defining reference levels of acceptable risk while keeping overall consistency with the well-established international standards for deterministic acceptance criteria.

Regulatory Standards are not prescriptive but of compliance with safety objectives (performance). The compliance of these objectives has to be demonstrated by the licensee by sound procedures within mandatory documents than can be objectively assessed by the Regulatory Body. The role of the latter is to be sceptical and critical, without imposing "how", which implies interaction between professionals of the Regulatory Body and the Licensee, in order to ensure a common understanding of the overall safety approach, including the statement of safety goals, the engineering solutions adopted, the analytical tools for proving safety and the methodology for deriving safety requirements.

ARN understands that performance based regulatory approach does not imply limiting the requirements to qualitative issues. Moreover, it is perfectly compatible with specific deterministic requirements and even numerical criteria. As examples, the Defence in Depth concept produces requirements on the independence of systems; the single failure criterion produces requirements on the need of redundancies for systems with components that may fail on demand; the requirement of using a conservative approach for the demonstration of Safety Cases (assessment of Design Basis Accidents against Safety Limits) is compatible with setting numerical requirements on safety margins. In brief, ARN approach is consistent with IAEA approach to the establishment of safety (engineering) requirements on functional capacity, reliability and robustness, derived from the safety classification of Structures, Systems and Components, which in turn is based on the Safety Analysis demonstrating the functional safety of a design.

ARN understands that a prescriptive regulatory approach is related to the development of requirements going beyond functional safety, extending to the imposition of the use of specific design solutions already characterised on a previously consolidated Plant designs. This approach allows to define on advance the standards to use for qualifying the compliance of safety requirements on functional capacity, reliability and robustness, without needing to assert specific safety requirements.

The regulatory system adopted by Argentina has consolidated according to the obtained results in the course of time.

Regarding the adoption of a performance based regulatory approach, some advantages, learnt by the verified application experience are the following:

- The nature of the interaction between the Regulatory Body and the Licensee contributes to an early detection of possible non-compliances or deficient compliance with regulatory requirements (in early design stages), avoiding the increase in time and efforts in fulfilling such requirements in later phases of a project (fabrication or construction).
- The design solutions to comply with regulatory requirements come, in general, from the supplier (Nuclear Vendor) through the Licensee, that know in detail the installation and the system involved in.
- The establishment of safety objectives keeping openness to different design solutions, helps to manage projects from different vendors, i.e. Nuclear reactors with different safety approaches, while keeping coherence on the need of objective (factual) demonstration of the compliance with regulatory requirements.

3.7.2.3.2. Basic criteria of radiological and nuclear safety

The basic criteria, on which radiological and nuclear safety is supported, have been applied since long time ago and are coherent with ICRP and IAEA recommendations.

Furthermore the ARN has contributed to formulate recommendations issued by international organizations (such as the mentioned IAEA and ICRP), so that it is usual to find, in its own standards, concepts dealing with radiological and nuclear safety that appear in such recommendations.

In case of emergencies the ARN also applies criteria consistent with ICRP applicable recommendations. (See article 3.16.).

3.7.3. LICENSING SYSTEM

3.7.3.1. GENERAL ASPECTS

A basic aspect of the Argentine regulatory system is the approach adopted, in which the Licensee deals with the design, construction, commissioning, operation and decommissioning stages of the NPP, being completely responsible for the radiological and nuclear safety of the installation as well as for the physical protection and safeguards. This responsibility goes beyond the compliance of requirements stated in the Regulatory Standards.

The Regulatory Standards establish that the construction, commissioning, operation or decommissioning of a NPP shall not be initiated without the corresponding *authorization:* License, which has to be previously required by the Licensee and later, issued by the Regulatory Body. *Despite that there is a validity period for the commissioning and operation Licenses, in all cases* the validity of such Licenses is *always* subordinated to the compliance with the conditions stipulated in its articles *of terms and conditions*. There are conditions on operation issues including staff training and qualification, emergency preparedness, radiological issues on workers, emissions and waste, transport of nuclear and radioactive material, safeguards, security and communication of the Licensee towards ARN.

The non-compliance with any of the regulatory standards, conditions or requirements is enough reason for the Regulatory Body to suspend or cancel the corresponding License validity, according to the sanction regime in force.

3.7.3.2. LICENSING PROCESS

3.7.3.2.1. Nuclear power plant licensing

The regulatory system considers licenses for construction, commissioning, operation and decommissioning that establish the conditions that the Licensee must fulfil at each stage.

The Construction License is issued when regulatory standards and requirements of the siting, basic design and expected safety operation conditions have been complied with prior to start of this stage.

The applicable regulatory standards, consistent with international recommendations on the subjects, establish the safety criteria to be met in the design of the installation and define the timetable and type of mandatory documentation that must be presented together with the application for the Construction License (Regulatory Standard AR 3.7.1.).

Once the Construction License is requested by the Licensee, a continuous interaction between the constructor or operator of the future installation and the Regulatory Body is initiated. It is a dynamic process, as complex as the demands involved. It should be emphasized that the Licensee's capacity to carry out its responsibilities is evaluated starting from the construction stage.

The Commissioning License establishes the conditions for the approach to criticality, operation with increasing power up to its nominal value, as well as verifications and tests of the components, equipment and systems to determine whether they comply with the original design basis. To do so the Licensee must appoint an ad hoc Commissioning Committee of senior specialists, to evaluate the execution of the commissioning program and recommends on its continuation and adjustment (Regulatory Standards AR 3.7.1. and AR 3.8.1.).

The Operating License is issued when the ARN verifies that particular conditions, regulatory standards and specific requirements applicable to a specific installation are fulfilled. Such conclusion is the result of analysing the submitted documentation and detailed studies, as well as the inspection results carried out during the construction and commissioning together with the ad hoc Commissioning Committee recommendations.

The Operating License is a document by which the ARN authorizes the commercial operation of a nuclear installation under stipulated conditions, which must be fulfilled by the Licensee (Regulatory Standard AR 3.9.1.). The non-fulfilment of any of the imposed requirements without the corresponding ARN authorization could imply the application of commensurate sanctions reaching the Operating License suspension or cancellation.

After the issuance of the Construction License, the overall responsibilities on safety, safeguards and physical protection of the Licensee remain in place regardless the circumstances of licenses, even under suspension or cancellation.

At the end of its lifetime and under the Licensee's request, the ARN authorizes the ending of the NPP's commercial operation and issues a Decommissioning License. In this document, conditions for the NPPs safe dismantling are established, being the Licensee in charge of planning and providing the necessary means for its fulfilment (Regulatory Standard AR 3.17.1.).

The safety evaluations performed prior to issuing a NPP License include mainly aspects of the mandatory documentation, such as Safety Analysis Report (SAR), Quality Assurance / Management System programs, construction procedures, operation procedures, previsions for in-service inspections, maintenance program, etc. Besides, emergency plans are prepared in co-ordination with the corresponding National, Provincial and Municipal Authorities.

In 2010, the Regulatory Body faced the need to license an innovative reactor design (CAREM reactor). For this purpose ARN defined an "ad hoc" licensing scheme based on the authorization of "non-routine practices". This licensing scheme foresees the following authorizations: for Use of Site and Construction, for Fueling, for Core Subcritical Testing, for initial criticality, for Zero Power Tests, for Power Increase and for Full Power Tests.

Later, ARN decided to analyze the "ad hoc" licensing scheme developed at an early stage in the project, and concluded there is a need to update and adapt it according to the requirements of AR standards (AR 0.0.1 and AR 3.7.1.) and the experience gained in other projects. The revised licensing scheme is similar to those applied for new NPPs in terms of requisites to be fulfilled by the Licensee. The scope of this modification will take place from the next licensing milestone, when the construction and preliminary tests are completed, establishing a Commissioning License.

3.7.3.2.2. Periodic Safety Review and Renewals of License

Until 2003, the Periodic Safety Review (PSR) as defined by IAEA had not been required in Argentina. Regulatory Standard AR 3.9.1. establishes that the SAR of NPPs must be updated each time that a plant design modification is performed and once every five (5) years. The NPP´s Operating Licenses include similar requirements. However, the Operating Licenses didn't have a defined validity period but they could be revoked when the utility didn't fulfil some requirement contained in the Licenses.

Those *integrated* safety reviews, which are part of the continuous improvement program, foresee a continuous follow-up of the safety conditions, the operative experience feedback and the Aging Management Program. Furthermore it is a regulatory requirement to perform and to update the NPP's PSA, which implies that a safety review be performed during the revision stage, or improvements implementation, or design changes.

However, in 2003 the Board of Directors of the ARN nominated a specific committee to analyze and update the NPPs Operating Licenses considering national and international aspects. A number of considerations were taken into account such as:

- limited or unlimited renewal period of the Operating License;
- avoiding repetitions of the regulatory standards;
- the benefits of applying a PSR methodology;
- the requirements issued that consider permanent fulfilment, and
- the operating experience.

The committee issued in August, 2003, a new draft of the NPPs Operating License that includes two major changes:

- To include a validity period of the Operating License of 10 years.
- To require a formal PSR for its renewal.

At the end of 2003, the Board of Directors of the Regulatory Body approved the committee's document and these major changes were put into practice in the new License for CNA I in 2003 and in the new License for CNE in 2007.

An exception was made in 2016 for CNA II licensing through a decision of Board of Directors based on the particular conditions of this delayed project (recall that CNA II Construction License dates from 1983, and the project was halted in 1994). According to that decision, the Operation License of CNA II was issued with a validity period of five years.

As above stated the validity of the Operating License is 10 years, and the continued operation must be justified by the submission of comprehensive PSR whose scope is defined by ARN in agreement with the Licensee. The approach for long term operation is also based on the PSR according to the latest IAEA SSG-25. The Integrated Implementation Plan for the definition of a safe long term operation program is derived from the global assessment of all the safety factors and categorized according to the safety significance. ARN stress the need to assess the cumulative effects of all plant modification as well as the effects of plant ageing and site-specific and organizational aspects. The decision for a safe continued operation is based on the remaining risk after the Integrated Implementation Plan's activities were performed. There is no "life" concept; instead Argentina follows the concept of acceptable remaining risk for a safe continued operation.

3.7.3.2.3. Nuclear power plant personnel licensing

Regulatory Standards AR 0.11.1. and AR 0.11.2. set the criteria and procedures to provide Individual Licenses and Specific Authorizations to the personnel who apply for licensable functions in nuclear installations. These regulatory standards also establish terms and conditions according to which the ARN may issue these Individual Licenses and Specific Authorizations.

In addition, Regulatory Standard AR 0.11.3. establishes criteria on retraining of personnel for this type of installations by means of specific requirements that have to be met for the plant staff training and the mechanisms for the evaluation of the training process. These requirements are met through courses and practices in order to keep the knowledge and the skills necessaries for the effective performance of the plant staff duties which include accidental situations. The corresponding control and monitoring are periodically performed by the Regulatory Body.

Two kinds of conceptually different documents, which imply certifications, are issued:

- Individual License: it is a certificate of permanent nature recognizing the technical-scientific qualification necessary for a person to perform a certain function within the operation chart of a certain type of nuclear installation. The Individual License is a necessary but not a sufficient condition for holding a licensable position in a given NPP.
- Specific Authorization: it qualifies a licensed person to perform a specific function in a particular nuclear installation. It has a maximum validity of two years and may be renewed after some conditions are met.

Whenever an Individual License or a Specific Authorization is needed for its personnel, the Licensee submits the necessary documentation to the ARN. The "Consejo Asesor para el Licenciamiento del Personal de Instalaciones Clase I y Clase II y III del Ciclo de Combustible Nuclear" (CALPIR - Advisory Committee for the Licensing of Major Installation Personnel), which advises the Board of Directors of the ARN concerning these matters, evaluates each applicant's qualification, and either suggests the issue of the requested certificate, or otherwise produces a requirement to the Licensee for the applicant's additional training so as to achieve the needed qualification.

The applicant for an Individual License or a Specific Authorization or for the renewal of the latter must fulfil a number of requisites concerning qualification, working experience, training, retraining and psychophysical aptitude, depending on the installation and on the function. These requisites may be summarised as follows:

To obtain an Individual License, it is required:

- Basic qualification: an education level (secondary, tertiary or post-graduate) suitable to enable the access to higher stages of qualification according to the technical scientific aptitude required considering the type of task and function level.
- Specialized qualification: the technical-scientific knowledge in the nuclear field required to perform a licensable function adequately. The specialised qualification applicant must attend training courses of the programs accepted by the ARN and pass examinations, which are overseen by ARN personnel.
- Working experience: significant experience for the correct performance of the function applied for.

To obtain or renew a Specific Authorization, it is required:

- Specific qualification: knowledge regarding radiological safety, installation procedures and characteristics, responsibilities of the position to be licensed and the mandatory documentation. The extension and depth of the applicant knowledge shall be such that it can contribute to the safe operation of the installation.
- On-the-job-training: to have carried out tasks corresponding to the function applied for, under the supervision of licensed personnel, in the same or a similar installation.
- Re-training: to take courses and periodic evaluations prepared by licensed personnel in the operation of a nuclear installation, with the purpose of updating knowledge and developing aptitudes which will enable the applicant to face eventual abnormal situations.
- Psychophysical aptitudes: the applicant psychophysical conditions shall be compatible with the psychophysical profile needed to perform a licensable function correctly. A physician proposed by the Licensee and recognized by the Regulatory Body establishes the psychophysical profiles and must certify the psychophysical aptitude of the applicants.

The Specific Authorization is obtained after taking courses according to programs accepted by the ARN, on-the-job-training finalization and passing examinations overseen by its personnel.

In the case of CNE's refurbishment outage, a specific retraining program was required by the Regulatory Body, in order to guarantee the maintenance of the operator skills, as well as to acquire the knowledge and operating changes according to the new design improvements of the unit, during the outage.

3.7.3.3. REGULATORY INSPECTIONS AND AUDITS

From the beginning of nuclear activity in the country, the Regulatory Body has performed, *as core functions, review and* assessments as well as multiple and different regulatory inspections and audits as frequently as considered necessary, with the purpose of verifying that nuclear installations satisfy the regulatory standards, Licenses and requirements in force. All these activities are performed according to written procedures.

Act No. 24,804, entitles the ARN to carry out with such inspections and regulatory *review and* assessments, performed by its personnel such as:

- Routine *planned* inspections are carried out by resident inspectors and other ARN personnel. Their purpose is to verify that the Licensee complies with limits and conditions of operation established in the mandatory documentation.
- Special inspections *including reactive inspections* are carried out by ARN specialists (dosimetry, instrumentation and control, thermo hydraulics, etc.) in coordination with resident inspectors. These inspections are performed under special circumstances or due to the occurrence of abnormal events in the installation.
- Safety Assessments are performed by ARN personnel and consist of the analysis of data obtained during inspections or any other source, for instance, radiological safety evaluations carried out during certain practices at the NPP, in order to identify eventual weak aspects or identify possible ways of reducing personnel doses.
- Regulatory Audits are planned and carried out by ARN personnel to analyze organization, operation and process aspects related to radiological and nuclear safety in order to examine the degree of compliance with the provisions in the mandatory documentation.

3.7.3.4. REGULATORY ACTIONS

The regulatory actions that the Regulatory Body may take in relation with a particular installation are originated mainly as:

- The results of regulatory *review and* assessments, inspections and audits carried out in the installation.
- The knowledge obtained from abnormal event occurrences in the installation itself or applicable events in other installations.
- The results of ARN technical evaluations.
- The application of recommendations or good practices arose from exchange of information, technical documents or lesson learnt coming from domestic or international sources.

In such cases, the ARN sends a regulatory document to the Licensee, which takes the form of a requirement, a recommendation or an additional information request according to the case. The document demands the Licensee to carry out the required corrective actions in a *timely manner according to the safety significance*.

Such documents have the following scope:

- Requirement: it is a regulatory demand that must be fulfilled by the Licensee as requested.
- Recommendation: it is a demand that differs from a requirement in that the Licensee has certain flexibility to accomplish it by means of alternative ways (e.g. engineering solutions), which ensure at least the same result required by the recommendation. Such proposals must be submitted to the Regulatory Body for evaluation.
- Additional Information Request: it is a regulatory demand in which a higher degree of details in the submitted documentation is requested; e.g. justification of certain assertions, demonstration of some calculation results, or additional documentation.

3.7.4. SANCTIONS REGIME

Non-compliance with the Regulatory Standards and requirements set out in the respective licenses or permits entitles Regulatory Body to impose the appropriate enforcement within the Sanction System. The National Nuclear Law Act No. 24,804 provides in article 16 e), f), g), h) and i) an enforcement regime which establish the following competences for the ARN:

e) Proposing to the Executive Power the transfer, extension or replacement of a concession for the use of a State-owned nuclear facility whenever there are elements advising to do so, or its expiration when based on non-compliance with the rules it issues with regard to radiological and nuclear safety.

f) Bringing civil or criminal lawsuits at the competent courts when there is non-compliance from licensees or authorization or permit owners ruled by this Law, as well as requesting for search warrants and for the aid of the police when such actions are deemed necessary for exercising the faculties granted by this Law.

g) Applying sanctions, that shall be graded on the basis of the severity of the infringement, such as warnings, fines to be applied proportionately to the severity of the fault and as a function of the potential damage involved, the suspension of a license, permit or authorization or their revocation. Such sanctions shall be appealable only for the purpose of remand before the National Administrative Contentious Court of Appeals.

h) Establishing procedures for the application of sanctions for the violation of rules issued while exercising its competence, while ensuring the principle of due process of Law.

i) Decide the seizure of nuclear radioactive materials, as well as the preventive closure of facilities subject to regulations of the Nuclear Regulatory Authority, when they lack the due license, permit or authorization, or when gross negligence is detected with respect to the compliance with radiological and nuclear safety standards or with the protection of facilities.

In this context, gross negligence means the acts involving a serious threat to the safety of the population or to the environmental protection, or whenever the application of physical protection or safeguards measures cannot be guaranteed.

For these purposes, through the Executive Decree No. 1,390/98 article 16 was regulated and the ARN was authorized to lay down the relevant procedures that may apply in case of violation of the standards to be issued in the exercise of its competence, ensuring the constitutional guarantees of due process and the defense rights.

In the case of Nuclear Power Plants and its staff there is a specific Enforcement Regime “Sanctions Regime for Nuclear Power Plants”, approved by ARN´s Resolution No. 63/99, which provides sanctions not only in the case of non-compliance with the regulatory standards but also in the case of non-compliance with the Regulatory Requirements, Mandatory Documentation and Terms and Conditions from the License.

The Sanctions Regime represents the last link of the safety chain. The ARN considers that if the regulatory system is really effective and the Responsible Organizations fully exercise their responsibilities, the application of sanctions and fines should occur only in exceptional cases. In this sense, additional ARN function is to make Responsible Organizations and Primary Responsible aware of their responsibility regarding safety, in order to increase safety culture through all levels of the organization structure.

3.7.5. COMPLIANCE WITH THE OBLIGATIONS IMPOSED BY THE CONVENTION

In Argentina, a legislative framework has been established and maintained to govern the nuclear installations safety. This framework provides:

- An appropriate set of Regulatory Standards to be applied in safety subjects.
- A licensing system.
- A *review and* assessment and inspection system to verify compliance with the mandatory documentation.
- *An enforcement system through* the sanction regime to be applied in case of non-compliance with mandatory documentation.

Therefore, the country complies with the obligations imposed in Article 7 of the Convention on Nuclear Safety.

3.8. ARTICLE 8: REGULATORY BODY

1. *Each Contracting Party shall establish or designate a Regulatory Body entrusted with the implementation of the legislative and regulatory framework referred to in Article 7, and provided with adequate authority, competence and financial and human resources to fulfil its assigned responsibilities.*
2. *Each Contracting Party shall take the appropriate steps to ensure an effective separation between the functions of the regulatory body and those of any other body or organization concerned with the promotion or utilization of nuclear energy.*

3.8.1. FUNCTIONS AND COMPETENCE OF THE REGULATORY BODY

The current functions and competence of the Nuclear Regulatory Authority (ARN) are established in the Nuclear National Law, Act No. 24,804 and its regulating Decree No. 1,390/98. The ARN acts as an independent agency under the jurisdiction of the Argentine Presidency and is subject to a public control system. As provided by article 7 of the Act No. 24,804 the ARN, is responsible for the regulation and control of the nuclear activity on matters of radiological and nuclear safety and security, as well as the control of the use of nuclear materials, licensing and supervision of nuclear facilities and international safeguards.

The above stated Law sets out that the regulation and control of nuclear activities are "subject to national jurisdiction". The ARN also acts as an advisory body to the National Executive Power in matters of its competence.

Although the duties assigned to the Nuclear Regulatory Authority as per Article 7 of Act No. 24,804 do not cover the control of the x-ray generators, which is under the scope of responsibilities of the Public Health Authorities of the National State, the Provinces and the Government of the Autonomous City of Buenos Aires, in accordance with Act No. 17,557.

Article 8 established the scope of the regulation as stated in this Law in order to:

a) Protect human beings from harmful effects of ionising radiation.

b) Ensure that nuclear activities carried out in the Argentine Republic comply with radiological and nuclear safety requirements.

c) Ensure that nuclear activities are not to be performed for purposes other than those authorised by this Law, and that regulations issued in the future comply with international commitments and Argentina's policy on non-proliferation of nuclear activities and

d) Prevent intentional actions that could lead to severe radiological consequences or to unauthorized withdrawal of nuclear materials or other materials, or equipments subject to regulation and control, as stated in this Law.

The Law establishes that the ARN financial resources shall be made up with regulatory rates and contributions from the National Government.

Law No. 24,804 assigns a wide set of faculties and responsibilities to the ARN, article 16, establishes duties, attributions and obligations, some of them are the following:

- Lay down the regulatory standards for nuclear and radiological safety, physical protection and surveillance of the use of nuclear materials, licensing and supervision of nuclear installations, international safeguards and transport of nuclear materials in respect of nuclear and radiological safety and physical protection.
- Grant, suspend, and cancel licenses for the construction, commissioning, operation and decommissioning of nuclear power plants.
- Grant, suspend and cancel licenses, permits or authorizations for uranium mining and milling facilities, safety of research reactors, significant accelerators, major radioactive installations, including installations for radioactive waste management and nuclear applications to medical and industrial activities.
- Undertake inspections and regulatory assessments at the nuclear installations that are subject to ARN regulations, with the regularity that may be deemed necessary.
- Impose sanctions, which shall be quantified according to the importance of the fault, and which may imply even the seizure of nuclear or radioactive materials, the preventive closure of the

installations if nuclear activities are performed without the appropriate license, permit or authorization or if non-compliance of nuclear and radiological safety and physical protection of materials and nuclear installations standards are detected.

- Establish, in accordance with international parameters, nuclear and radiological safety standards for the personnel working at the nuclear facility and grant the specific licenses, permits and authorizations to perform the function subject to license, permit or authorization.
- Assess the environmental impact of any licensed activity, including monitoring activities, review and follow-up of said impact, and evolution or possibility of harm to the environment as a result of the licensed nuclear activity.

Thus, Law No. 24,804 *and Annex I of regulatory Decree No. 1,390/98* grants to the ARN the necessary legal competence to establish develop and implement a regulatory and supervisory system for nuclear activities performed in the country. In order to ensure an appropriate control, said legal competence is supplementary to adequate technical competence. That is, the ARN has the capacity to evaluate by itself the design, construction, commissioning, operation and decommissioning of monitored nuclear installations.

Since the beginning of regulatory activities in the country, it was imperative to have qualified personnel with knowledge experience and independent criteria in all aspects of nuclear and radiological safety, safety in the transport of radioactive materials and in radioactive waste management, safeguards and physical protection.

In compliance with the provisions of Law No. 24,804 and its regulating decree, the ARN has created the Nuclear Emergency Intervention System (SIEN in its Spanish acronym), complemented by the SIER (Radiological Emergency Intervention System) intended to work in the organizational frame of the Federal Emergency System (SIFEM in its Spanish acronym, created by Decree No. 1,250/99). The ARN functions related to Emergency Preparedness are explained in Section 3.16. of this Report.

Law No. 25,018/98 sets provisions that involve the ARN in the management of Radioactive Wastes. It states that ARN must:

- Approve the acceptance criteria and the transference conditions of the radioactive waste formulated by CNEA (as application authority).
- Approve radioactive waste transference procedures, as well as irradiated fuel elements established by CNEA (CNEA, in this case, as the radioactive waste generation centre).
- Advise the National Congress in relation to the Radioactive Waste Management Strategic Plan.

The ARN also takes appropriate measures consistent with its national law to protect the confidentiality of the information. This provision is considered in article 16 j) and in Resolution ARN No. 67/04 along with the provisions established in Decree No.1,172/03.

3.8.2. ARN ORGANIZATIONAL STRUCTURE

According to the provisions in Law No. 24,804, ARN is managed and administrated by a Board of Directors, all members with adequate technical and professional background on the subject. The Board of Directors consists of three members, a Chairman and 2 Vice-Chairmen.

ARN acts as an autarchic organization and depends directly of the Executive Power through the General Secretariat of the Presidency.

The ARN organization is matrix based, where the different tasks involving different sectors are designed as projects or activities, for a better use of the available economic and human resources.

Activities are permanent tasks along years (i.e. regulatory inspections) while the projects have a limited duration and, once they are completed should be integrated into one or more activities. A schematic chart of the ARN organizational structure in force since 2010 is shown in Figure 3.8.1.

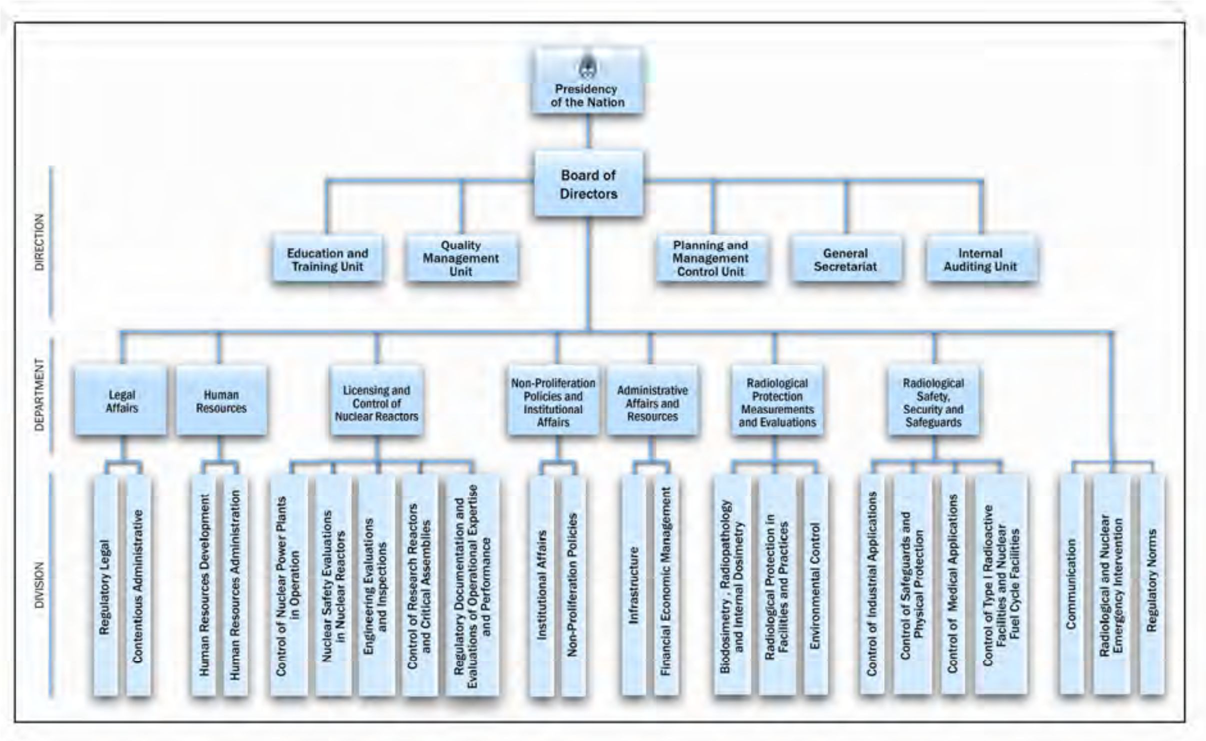


Figure 3.8.1. – ARN Organization Chart

The General Secretariat as well as the Units of Internal Auditing; Planning and Management Control; Quality Management; Education and Training (UCE, Unidad de Capacitación y Entrenamiento) reports directly to the Board of Directors.

The General Secretariat has the main function of assisting the Board of Directors on the managerial and administrative matters and is also the manager of some common infrastructure sectors as IT, information registry, cafeteria and internal communication.

The Unit of Planning and Management Control reports to the Board of Directors on the distribution and prospective of resources use. This allows the follow-up of the strategic plans and management control about the use of financial resources of projects and activities.

The Quality Management Unit has the responsibility of manage the ARN Quality Management System.

The UCE has the responsibility for implementing and maintaining the quality of the Postgraduate Educational Courses (PGEC); the preparation of analyses on strategies and action courses for the education and training of the ARN staff and the institutional knowledge management.

The Radiological Safety, Security and Safeguards Department carry out regulatory assessments and inspections concerning radioactive installations (medical, research and industrial installations), transport of radioactive and nuclear materials, safeguards control, and physical protection controls. Besides, this department controls the use of nuclear materials, equipment and installations of nuclear interest verifying the compliance with international agreements related to non-proliferation guarantees.

The Radiological Protection Measurements and Evaluations Department performs specialized evaluations on radiological safety involving people and the environment, by carrying out several measurements and modelling of different scenarios.

The Administrative Affairs and Resources Department provide administrative and accounting support to ARN’s regulatory tasks.

The Non-Proliferation Policies and Institutional Affairs Department elaborates proposals in the area of nuclear policy under the purview of ARN, and implements the institutional policies established by the Board of Directors. Furthermore, this department coordinates the institutional relations in the national and international sphere.

The Licensing and Control of Nuclear Reactors Department, is in charge of the control and licensing of NPPs (CNA I, CNA II, CNE, CAREM and Fourth NPP –see section 3.8.3.1.) and research reactors. Its activities include assessment of the design, construction, commissioning and operation stages, as well as the corresponding inspections and regulatory audits.

The Human Resources Department is responsible for the development and administration of the ARN human resources in regard of hiring personnel, managing salaries, organising promotions, implementing scholarships and internships.

The Legal Affairs Department participates in the assessment on legal aspects of ARN activities, such as purchasing processes, contracting, licensing and penalties.

The Regulatory Norms, the Radiological and Nuclear Emergency Intervention and the Communication Divisions, reports directly to the Board of Directors.

The Division of Regulatory Norms is responsible for issuing and establishing the Regulatory Standards and Guides, which regulate and control nuclear activities along the whole national territory (see section 3.7.2.).

The Radiological and Nuclear Emergency Intervention Division is responsible for develop the functions of ARN related to nuclear emergencies, preparedness, training and response (see article 3.16.).

The Division of Communication is responsible for promoting the ARN's institutional image among the stakeholders through strengthening internal and external communications (see section 3.8.5.).

3.8.3. ARN HUMAN RESOURCES

The current ARN's staff is formed by 361 personnel. Nowadays, 59% of 353 ARN personnel are assigned to core regulatory activities belonging to the processes of licensing and control of nuclear facilities: review and assessment, inspection and audits. The remaining 41% are assigned to support activities.

Regarding the level of education, it is stressed that 59% of ARN personnel have university degree, either master degree or Ph.D.

The distribution of professional staff is the following: 37% belongs to engineering branch, 28% to Natural Sciences area (physics, chemistry, biochemistry, environmental and medicine), 11% is specialized in social sciences, 7% belong to economic sciences, 3% to legal area and the remaining 14% belongs to other specialities.

In relation to engineering branch the ARN is composed of 27% Electronic, 6% Electric, 3% Nuclear Physicians, 29% Chemical, 3% Civil, 10% Industrial, 11% Mechanical and 11% to other specialities.

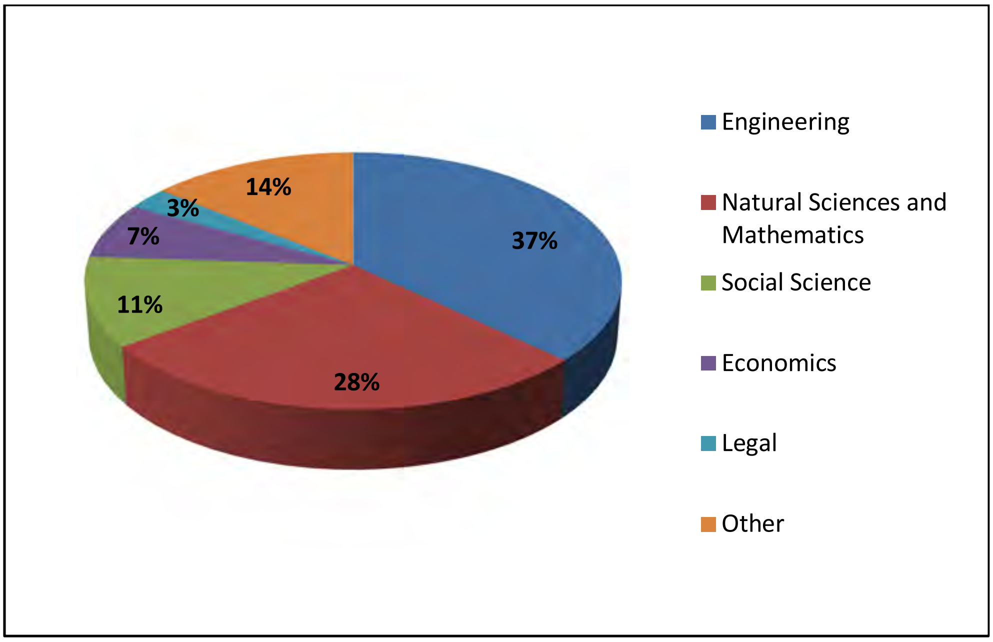


Figure 3.8.2. – ARN professional staff distribution

Presently ARN counts with 9 advisers, with the aim of retaining the talents of retired professionals.

ARN staff is geographically distributed as follows: 74% at headquarter in Buenos Aires City, 22% in Ezeiza Atomic Centre, 3% in Nuclear Power Plants and 1% in Ushuaia and Bariloche.

In addition, there are ARN experts temporarily performing functions in international organizations in the following way: 3 persons in the Argentine-Brazilian Agency for Accounting and Control of Nuclear Material (ABACC); 3 in the IAEA and 1 in WHO (World Health Organization).

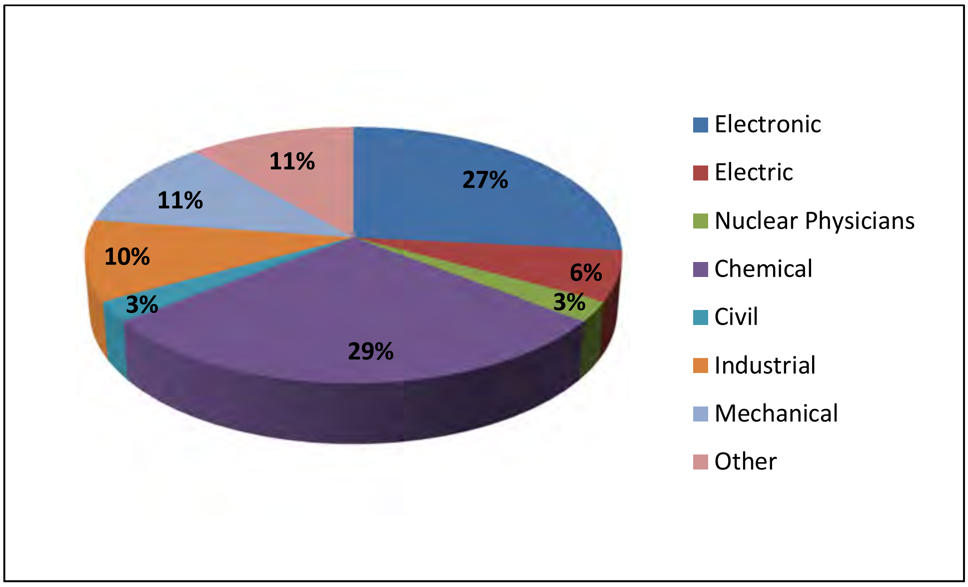


Figure 3.8.3. – Distribution based on engineering branches

The Natural Sciences area is composed of 32% physics, 20% chemistry, 20% biochemistry and biologic science, 9% environmental, 7% medicine and the remaining 12% belongs to other specialities.

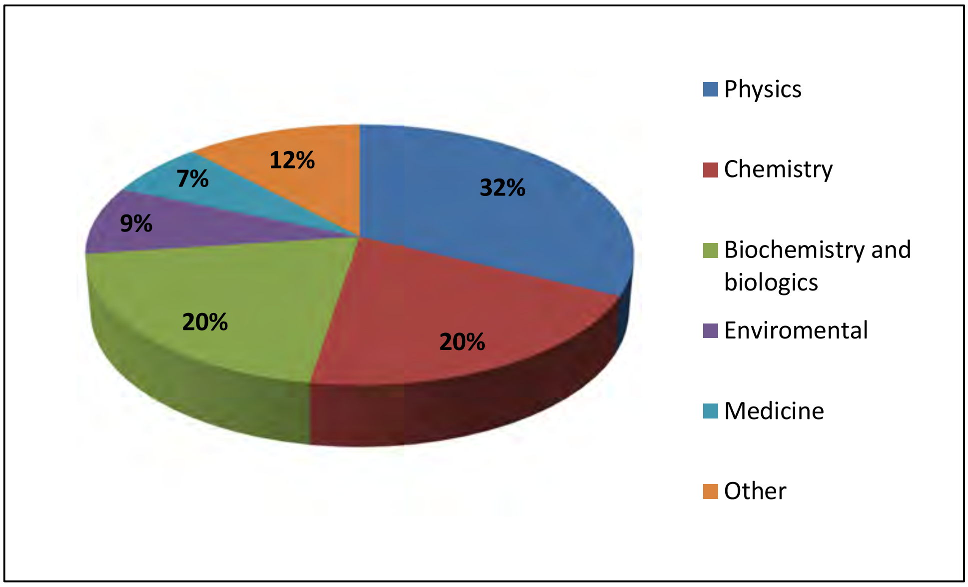


Figure 3.8.4. – Distribution based on natural sciences disciplines

In the period reported, there have been activities in order to define the ARN job profiles. The Human Resources Development personnel will develop a training plan for the medium and long term.

3.8.3.1. RESOURCES ASSIGNED TO THE NUCLEAR POWER PLANTS REGULATORY CONTROL

3.8.3.1.1. General aspects

In Argentina, the nuclear energy renaissance occurred in 2006, produced a positive impact in personnel motivation that helped improving ARN's staff situation, by offering to the professionals the opportunity of participation in attractive technological projects. This positive situation was slowed down after Atucha II's licensing project was finished and concurrently, the new NPP's licensing project suffered a delay in its commencement.

In addition to the above mentioned the country's economic situation during the reported period doesn't provide a good environment for hiring new personnel or for giving to the existent staff more attractive work conditions. Nevertheless, ARN continues fulfilling its responsibility with its own staff in conjunction with the use of Technical Support Organizations (TSO).

Management of activities with the existent plant personnel is possible as far as a graded approach for the planning is used. So, having a well define planning is one of the key elements for ARN´s management of activities.

Other key element is related to the knowledge of the personnel. In this sense, ARN undertakes, as a permanent activity, training of specialists in radiation and nuclear safety, safeguards and physical protection, by means of training courses and the participation in local and international training activities. These activities are carried out through the UCE in charge of defining, organizing and coordinating courses, workshops and follow-up seminars.

Young recruited professionals have assisted the Postgraduate Course in Radiation Protection and Nuclear Safety organized in agreement with the University of Buenos Aires (UBA) and the Ministry of Health under the IAEA auspices. This course took place uninterruptedly on a yearly basis from 1980 to 2004.

In 2005 the original course was divided into the following two yearly postgraduate courses:

- “Radiation Protection and Safety of Radiation Sources”. The course lasts 25 weeks, with a daily seven-hour work-load.
- “Nuclear Safety”. The course lasts 10 weeks, with a daily seven-hour work-load.

Additionally, a course in radiation protection is provided to train technicians. This course spent eight weeks with a daily seven-hour work-load.

After this basic instruction, the junior recently recruited professionals and technicians participate in on the-job training, collaborating in specialised technical tasks.

In addition, different ARN groups provide specific courses, as needed, and some junior and senior professionals are sent to attend specific courses in national or international institutions to improve their knowledge on special matters of ARN interest.

Joint tasks developed by ARN in collaboration with different TSO, are also used as an opportunity to incorporate knowledge and experience by junior and senior professionals.

As can be seen in section 3.8.7., ARN has several agreements with local and international support institutions. One of the conditions of the agreements is that the institutions involved do not provide assistance to the licensees in the same area.

The international institutions were selected for their expertise in the field concerned, for being well established and recognized, and also for their vast experience in advising other Regulatory Bodies.

Among these institutions, it is possible to mention TÜV and GRS from Germany and Sandia National Lab from USA and TECNATOM from Spain. On the other hand, the Argentinean company INVAP (regulatory branch), the Litoral University and San Juan University are regular TSO on safety assessment issues. On a related field, there is an agreement with the US-NRC on the use of computer codes for the modelling and assessment of nuclear reactors.

ARN trains its own personnel as well as technical personnel from other national and international institutions on radiological protection and safety of radioactive and nuclear sources (see section 3.8.3.1.5.).

In summary, ARN has a reasonable compliance of the requirements of resources and infrastructure to carry out its mission and to reach its objectives.

3.8.3.1.2. Human Resources assigned to operation control

According to the ARN organization the Licensing and Control of Nuclear Reactors Department (LCNRD) is in charge of the NPPs control and licensing (see Figure 3.8.5.).

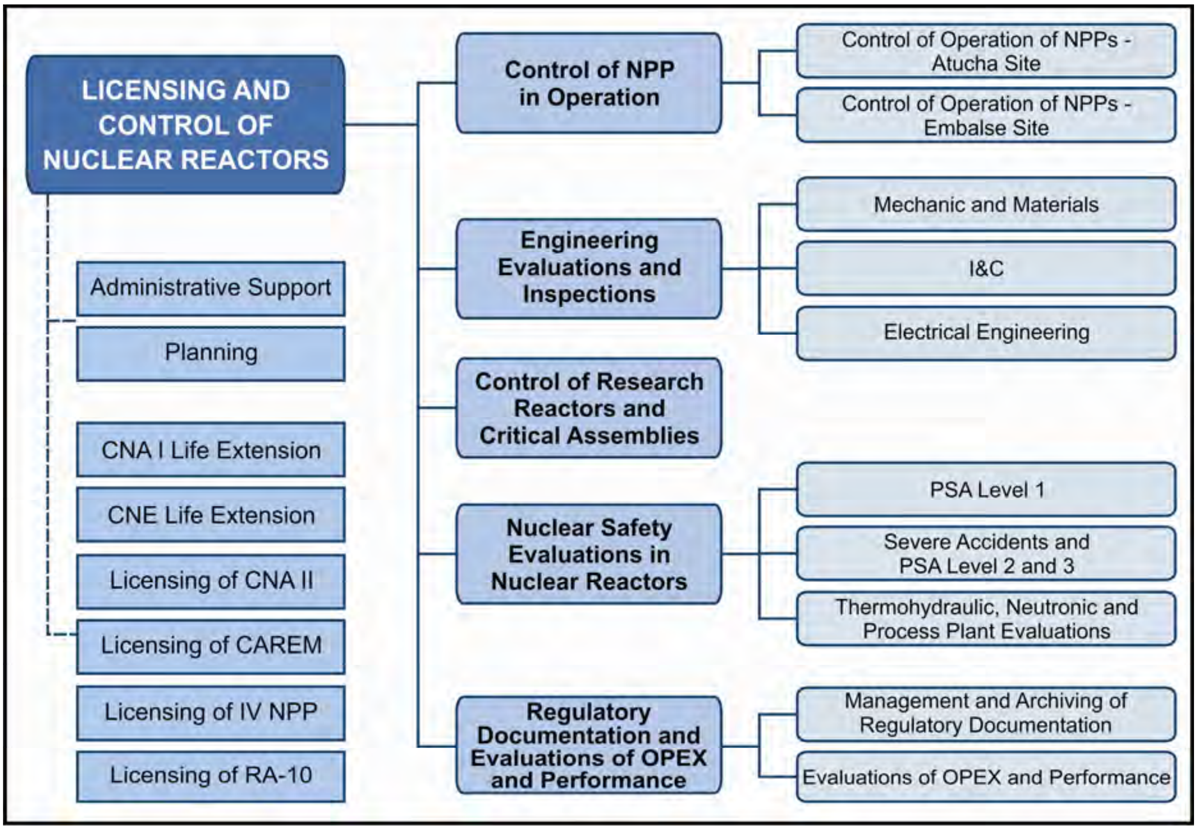


Figure 3.8.5. – Structure of the Licensing and Control of Nuclear Reactors Department

About *60* people are involved in regulatory activities related to NPP control and licensing. The percentage distribution of human resources assigned directly to NPPs inspections and safety assessments corresponding to *2019* is the following:

Inspections and evaluations in NPPs	*75%*
Support activities directly related to safety	*18%*
Support activities indirectly related to safety	*7%*

These tasks include resident inspectors in each NPP, and the safety analysers, who perform supporting tasks to inspection activities and study particular issues regarding the installation safety.

In addition, ARN elaborates and execute an annual Regulatory Audits Plan that involves areas of regulatory interest of CNA I, CNA II, CNE and CAREM to verify compliance with regulatory standards, especially the regulatory standard AR 3.6.1. (Nuclear Power Plant Quality System), as well as all other applicable mandatory documentation. Audit teams are composed by qualified auditors belonging to LCNRD and other areas of ARN.

The structure of the Licensing and Control of Nuclear Reactors Department was modified in 2015. One of the main changes affecting NPPs control was that inspections and evaluations related to radiological safety are carried out by Radiological Protection in Facilities and Practices Division belonging to Radiological Protection Measurements and Evaluations Department (See Figure 3.8.1.), in coordination with but independent from LCNRD.

3.8.3.1.3. Human Resources assigned to commissioning control

The human resources of the Regulatory Body involved during CNA I's construction and commissioning stages (1969-1974) are different from those assigned to CNE (1976-1984) and to CNA II (since 1981) for the same stages. This is due to the different circumstances under which those activities were undertaken and to the Regulatory Body's different experience in such cases (ARN since 1994).

For CNA I the role of Independent Authorized Inspector, prescribed by the ASME code, was performed by two entities: Technischer Überwachungs Verein, Baden (TÜV), appointed by Siemens Company and Control e Inspección de Seguridad de Centrales Nucleares (CISIN - NPPs Safety Control and Inspection) on behalf of CNEA. These entities carried out verifications of components fabrication, functional test, preliminary tests, etc.

For CNE, in order to co-ordinate the tasks related to licensing during construction and commissioning, ARN organised a special committee called Executive Committee for CNE Licensing. The main functions of this Committee were to analyse design and commissioning documents and to carry out inspections, audits and assessments. This committee performed the safety assessments during the plant construction and commissioning on its own or by contract with third partners / TSO assistance. The CNE seismic re-evaluation was of special significance.

In the CNA II case, the Regulatory Body licensed a second-generation NPP whose Construction License was granted in July 1981. The construction has stretched over nearly three decades during which the Regulatory Body has analysed several times the licensing aspects of CNA II, and concluded that it was feasible as long as the pertinent regulations were complied. In 2006 ARN established an ad hoc organization for the commissioning of CNA II.

As it was mentioned in 2013 National Report, the functions of the mentioned ad hoc organization were included in the Licensing and Control of Nuclear Reactors Department.

Inspections and evaluations related to the tests executed during Phases A, B and C were carried out by LCNRD personnel supported by experts belonging to some of the TSOs mentioned before.

In the case of CNE Life Extension, the regulatory oversight of the commissioning activities for returning to service the plant was performed by LCNRD personnel. This included the oversight of the complete functional tests for individual, either new or refurbished, systems important to safety, as well as the integral nuclear commissioning tests.

3.8.3.1.4. Human Resources assigned to construction, refurbishment control

In order to plan regulatory activities and optimize human resources to control the construction of CAREM Reactor and CNE refurbishment, ARN defined a set of structures, systems and components in which regulatory effort is focused.

Regarding CNE refurbishment, personnel from LCNRD and other technical areas of the ARN has been *involved,* with the purpose of strengthening the inspection staff of the CNE.

To date construction tasks of CAREM Reactor have not required a major regulatory effort.

3.8.3.1.5. ARN personnel qualification

The scheme for the training process of ARN's regulatory staff begins from the process of search for new personnel. The search is based on a job profile defined by the sector which has promoted the searching. This profile, which takes part of the general job profiles of the institution, includes the tasks to be performed by the agent as well as the knowledge and experience requirements that are considered necessary for his adequate performance.

The new personnel must go through an initial assimilation period that allows him to understand the objectives and functions of the institution and the specificity of his tasks. This process includes an "induction course" designed to introducing the new agent into general concepts related to the regulatory practices and a coaching period guided by senior staff in his working area.

According to the search criteria, the new agents can be pre-graduate or university graduate. This background education determines the process of training of entrants which first step is focused to introduce them to an applied knowledge. In the case of the pre-graduates destined to the technical areas, they are asked to pass the "Basic Training Course on Radiological Protection". The university graduates must attend the "Specialization Degree in Radiological Protection" and/or the

"Specialization Degree in Nuclear Safety", postgraduate courses that the ARN dictates in academic partnership with the School of Engineering of the University of Buenos Aires (FIUBA).

A fundamental element of the training of ARN technical personnel is the On-the-job Training (OJT) that each agent performs in their working sector, under the supervision of one or more senior regulators. So far, the ARN does not have, formal OJT programmes but these are defined by each sector according to needs. The UCE is working to formalize OJT.

The training of its agents is complemented through the attendance to other training activities in the modalities of Specialized Training Courses, Fellowships, Scientific Visits, Seminars or Conferences in National and International Institutions.

While not being one of its specific legal obligations, it has been a permanent policy of the Nuclear Regulatory Authority to organize and deliver courses providing applied education and training to future workers and to train the trainers to create a cascade effect of the training effort.

Since 1980 two Post Graduate Educational Courses on Radiation Protection and the Safety of Sources and on Nuclear Safety, and one Basic Training Course on radiation protection for under graduates have been dictated by ARN specialists for attendees of Latin America, the Caribbean and even from other countries of the world.

In 2006, the IAEA's first EduTA mission (Education and Training Appraisal) in a Latin American country was carried out in Argentina. This international peer review of the national educational infrastructure on radiological safety, concluded with very positive results for our country.

As an outcome to the EduTA mission, in 2008, the Government of Argentina signed a Long-Term Agreement (LTA) with the International Atomic Energy Agency for which our country assumes the responsibility of becoming a Regional Training Center (RTC) for Latin America and the Caribbean for the education and training in Nuclear, Radiological, Transportation and Waste Safety in Spanish language. The UCE is in charge of managing this RTC.

In November 2017 an EduTA follow up mission was conducted in Argentina. The EduTA team noted that ARN has been playing a fundamental role in the region in building competence in radiation protection and safety. Additionally, they noted that the continuous conduct of the postgraduate courses in collaboration with FIUBA under the auspices of the IAEA for the last 37 years represents a unique case of long-standing partnership between the IAEA and an RTC in providing assistance to Member States to build competence.

During the period 2016 to 2018, 26 ARN professionals achieved the Specialization Degree in Radiological Protection and 16 ARN professionals the Specialization Degree in Nuclear Safety. Additionally, 44 professionals from Latin America and the Caribe got the Specialization Degree in Radiological Protection and 13 the Specialization Degree in Nuclear Safety. During the same period, 97 under graduate workers attended to the "Basic Training Course on Radiological Protection", 21 of them were ARN staff.

Other activities managed by the UCE, during the *2016 to 2018* period, related to the capacity building of the regulatory staff were:

- *The organization of specific internal courses required by different ARN Divisions and Departments.*
- *The organization of tailor- made courses required by stakeholders.*
- *The participation in the Steering Committee Meetings.*
- *ARN experts also take part in safety related activities in organizations such as ICRP, UNSCEAR and IAEA (e.g. CSS, INSAG, IRS, NSGC, NUSSC, RASSC, TRASSC, WASSC) in the role of consultants and also attend technical meetings. They also participate in OSART, IRRS and in international expert missions. Each activity developed allows a fruitful exchange of experience and lessons learnt and promote a high level of competence in safety.*

3.8.4. QUALITY MANAGEMENT SYSTEM IN THE REGULATORY BODY

ARN has established, documented and implemented a Quality Management System (QMS) according to the requirements established in the ISO 9001 Standard.

The actions and requirements fulfilment are described in the Quality Manual (MC-ARN)*, which was updated in 2018*. In this document the ARN Board of Directors declares and communicates the Institution's Quality Policy and Commitment.

QUALITY POLICY

ARN's commitment is to protect people, the environment and future generations from the harmful effects of ionizing radiation and to ensure that nuclear activity developed in Argentina is only for peaceful purposes.

ARN promotes a safety culture based on a system focused in raising awareness of the value of safety, following a prudent and rigorous regulatory approach and giving a transparent access of radiological and nuclear safety's aspects to all stakeholders.

ARN plans, does, check and acts for the continuous improvement of its Quality Management System, in accordance to its strategy goal

3.8.4.1. PROCESS APPROACH IMPROVEMENT

ARN implements its QMS with the purpose to continuously improve the efficiency and effectiveness in the regulatory functions always oriented in increasing the stakeholder's satisfaction.

The QMS is implemented by applying a process *oriented* approach. *By following this approach, the understanding and management of the interrelated process like a system, contribute to efficiency and effectiveness of the organization in fulfilment its goals.*

In order to forward the implementation of ISO 9001 2015 version, the Process Map was updated. (See Figure 3.8.6).

Sequence and interaction of these processes are established and represented in a Process Table.

Each process is described in a Process Table, where the objectives, responsible, *processes*, suppliers, inputs, outputs, activities, actions for non-conform product, evidences of product characteristics, performance indicators *and involved divisions*, among others, are considered.

3.8.4.2. MEASUREMENT, ASSESSMENT AND IMPROVEMENT

The assessment of the process's performance is done through different methods: internal audits, process follow up, process monitor, review by the high Direction, QA verifications, evaluation of indicators related with the fulfilment of process's objective and survey analysis.

ARN performs internal audits with the purpose of:

- *Demonstrating product and process conformance against the applied requirements.*
- *Assuring conformance of QMS.*
- *Improving continuously the efficiency of QMS.*
- *Identifying improvement opportunities in order to continuously improve the efficiency of QMS by using quality goals.*
- *Verifying that corrective actions are done and assessing their effectiveness.*

The internal audits are developed by qualified personnel, always belonging to an independent from the audited division.

During the reported period 32 internal audits were performed and the Instituto Argentino de Normalización y Certificación (IRAM) performed 13 external audits to the certified processes.

The process follow up is done through different meetings where the responsible and personnel involved in the process, are present. The objective of the meeting is to verify the use of management tools and the approach for the issues (non-conformances) found during the audits.

The high Direction (ARN Board) performs a QMS in depth review with a given frequency and introduce modifications, as far as necessary, in order to ensure its adequacy and efficiency.

Finally, quality verifications can be performed in order to collect information in relation to an activity or a set of them, with the purpose of checking the fulfilment of requisites as stated in the applicable documentation.

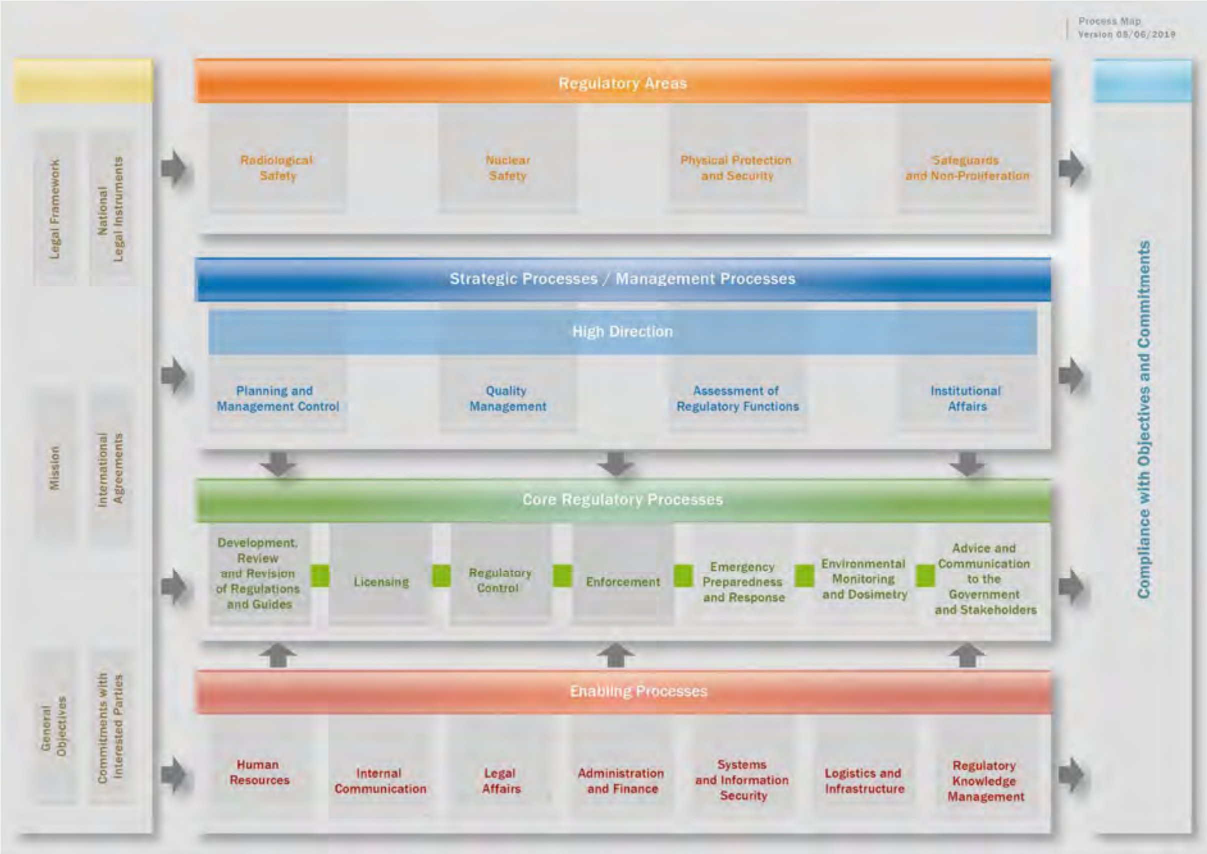


Figure 3.8.6. – ARN Process Map

3.8.4.3. CERTIFICATION AND ACCREDITATION

The certified processes up to August 2017 were 13. Later, and driven by budgetary reasons, the Board of Directors decided don't continue certifying processes, instead decided to continue with the mandate of processes adequacy to the latest version of ISO 9001 standard.

On the other hand, the Radiological Protection Measurements and Evaluations Department since 2016 has already accredited laboratories and some are under implementation process following the standard IRAM 301:2005 (ISO/IEC 17025:2005), "Requisitos generales para la competencia de los laboratorios de ensayo y calibración" and the criteria established by the Organismo Argentino de Acreditación (OAA).

Up to date, the Environment Control Laboratory (LE 116) and Biological Dosimetry Laboratory (LE 147) have completed satisfactorily the assessment for the fourth maintenance period under full scope of the standard (re-assessment). The Thermal-luminescence Dosimetry Laboratory has satisfactorily completed the assessment for third maintenance period under full scope of the standard (re-assessment) ending the first cycle of accreditation. Besides, the Calibration Laboratory has satisfactorily completed the assessment of the second maintenance period.

3.8.4.4. DOCUMENTATION REQUIREMENTS

The documental structure of the QMS, is made up of external documentation that is taken as a reference for the development of the ARN activities, being referenced in the internal documentation generated by the different processes of ARN.

In order to fulfil the documentation record requirements stablished in ISO 9001 standard and in the requirement defined by ARN for QMS efficiency, the following records were included in the processes:

The records associated are the basis and support of the implementation of activities.

During the reported period 109 documents (manuals, regulations, procedures and work instructions), 8 processes tables and 13 record control, were either done or updated.

3.8.4.5. SATISFACTION OF STAKEHOLDERS

ARN has into consideration the stakeholder's expectatives in all of its activities as well as in QMS processes interaction, with the main purpose of increasing stakeholder's satisfaction ensuring at the same time that safety is not challenged.

ARN Board of Directors ensure that all the stakeholder's requisites be known by the organization personnel and satisfied as far as possible, giving the first priority to the fulfilment of the Nuclear National Law, Act No. 24,804, protection of the population and workers, as well as the environment.

3.8.4.6. CONTINUOUS IMPROVEMENT

To improve its QMS, the ARN analyzes the data from internal quality audits, compliance indicators, management reviews, analysis of surveys, suggestions and opportunities for improvement coming from the stakeholders.

Based on the results of the analysis, it is determined if there are needs or opportunities that must be considered for continuous improvement.

Likewise, all those deviations (Non-compliance, corrective action) that are detected both internally and during an internal quality audit must be managed to make the corresponding corrections and take the appropriate corrective actions.

3.8.5. COMMUNICATIONS WITH THE STAKEHOLDERS

ARN has the legal obligation to inform the public and the willingness to communicate with stakeholders.

Regarding external communication, the objectives are to foster public understanding of ARN's role and activities; to work on the perception of risk by the public; *to develop informational materials that can be understood among different stakeholders, specially general public*, to manage conflict resolution in the national nuclear area and institutional crisis involving media issues; and to strengthen communications with other national and international institutions and stakeholders. In this sense, open channels of communication need to be constantly maintained.

ARN *annually* issues publications which are aimed at keeping stakeholders informed of the activities undertaken by the institution.

In ARN's Website regulatory information can be found, such as Regulatory Standards, laws and acts, permits and operating license information, environmental radiological monitoring, citizens' protection against incidents, technical reports on radiological and nuclear events, and general public communications among others.

In November 2015 ARN opened its official page in Facebook, the social network with more followers in Argentina (www.facebook.com/AutoridadRegulatoriaNuclear). The presence of ARN in social networks aims to open new channels of communication to reach with information on regulatory actions, important news of the nuclear industry, events and courses, and to promote more public spaces for citizen participation through messages, suggestions and comments.

The ARN's Strategic Plan 2018-2022 placed as one of its strategic objectives the improvement in access to information on the ARN functioning by stakeholders in general.

In 2018 ARN began with a survey of its communication channels in order to evaluate its accessibility and effectiveness.

In February 2019 the ARN website was relaunched at www.argentina.gob.ar/arn *pursuant to the Article 3 of the Decree No. 87/2017,.which establishes that all the Argentine public administration websites have to host in a centralized and unique State website. This opportunity enhanced a content reorganization, tailored to 3 different users' profile –general public, regulated and students, in a more modern and accessible website for people with different disabilities, and with responsive design.*

3.8.6. FINANCIAL RESOURCES

The effective fulfilment of the regulatory objectives requires that ARN has an efficient structure and adequate personnel together with the necessary economical resources. Concerning this matter, Law No. 24,804, establishes that such resources must be basically obtained from the following incomes:

- Annual regulatory fees.
- Supports from the National Treasury determined in each fiscal budget.
- Other funds, goods or resources that could be assigned through Acts or pertinent regulations.

In the case of NPPs in operation, the mentioned Law sets the amount of the annual regulatory fees, as a function of the nominal power installed for each NPP, which must be paid annually by the Licensee until the end of the tasks concerning the withdrawal of irradiated fuel elements from the core during decommissioning.

In the case of NPPs under construction, the fee which is set forth in the "Regime for the Regulatory Rate for New NPPs" is applied. The fee covers the cost of regulatory activities during the NPPs construction, erection, preliminary tests and commissioning stages.

ARN annually issues a budget proposal containing the detailed income previsions due to regulatory fees, and explains the request of funds from the National Treasury. This budget proposal is published in such a way as to clarify the expenses to be paid by people and institutions compelled to pay these regulatory fees.

The budget assigned to the ARN for the years 2016 and 2018 is shown in Table 3.8.1. The total budget during 2018 is composed of: 74% from the National Treasury, 24% of annual regulatory fees and goods or resources assigned according to applicable laws and regulations, and 2% of donations.

Table 3.8.1. – ARN comparison budget for financial years 2016 and *2018*

Item	*$ (in thousands of Argentine pesos)*	
	2016	***2018***
1 - Personnel	*208,695*	*294,595*
2 - Support goods	*10,448*	*11,770*
3 - Services	*102,006*	*73,492*
4 - Equipment	*16,136*	*38,986*
5.1 - Fellowships	*1,900*	*2,000*
5.9 - Transfers	*10,047*	*11,000*
9 - Other expenses	*4,516*	*7,550*
TOTAL	***353,748***	***439,393***

The ARN percentage budget distribution is shown in Figure 3.8.7.

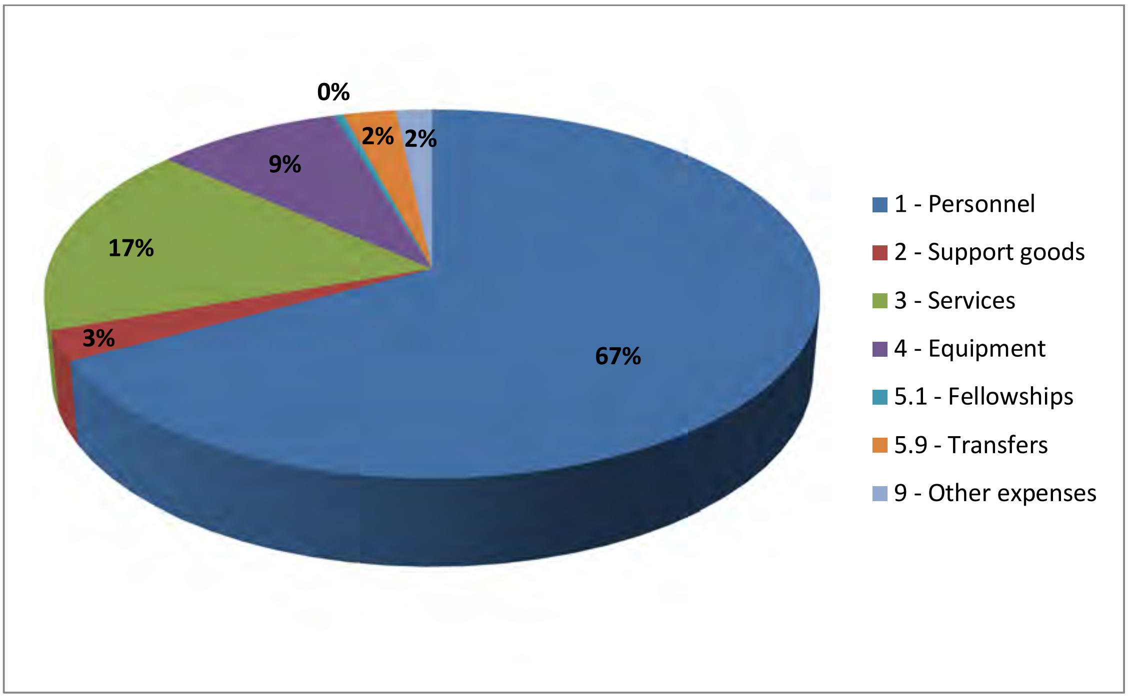


Figure 3.8.7. – ARN *2018* percentage budget distribution by category

Additionally, the financial resources distribution for 2018, by tasks and type of inspection, is shown in Figures 3.8.8. and 3.8.9.

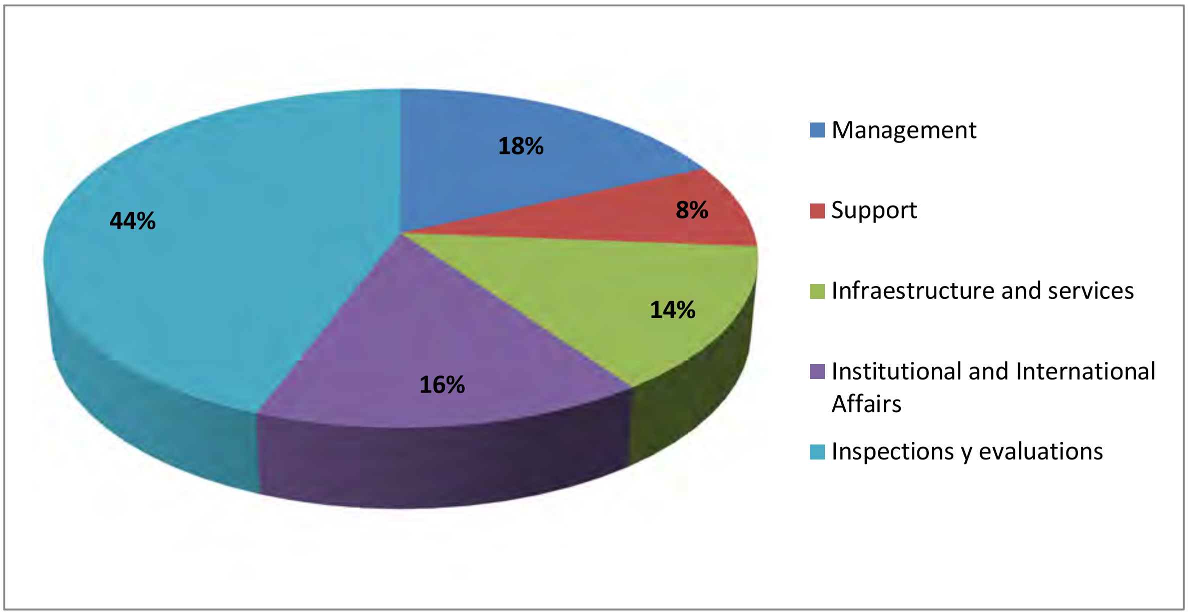


Figure 3.8.8. – Budget distribution by tasks

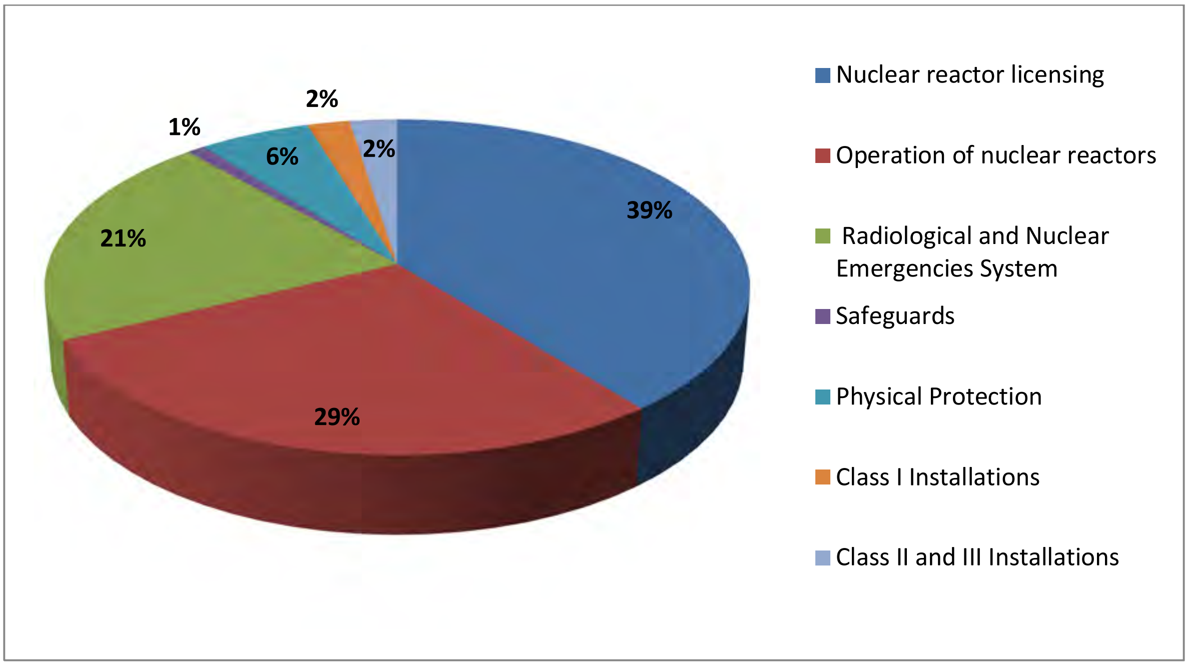


Figure 3.8.9. – Budget distribution by type of inspections

3.8.7. RELATIONSHIP WITH OTHER ORGANIZATIONS

Regarding to its functions, ARN keeps an active interaction with several national *and international*, governmental and private institutions, with the purpose of promoting experience and information exchange and developing technical co-operation with them.

In the period belonging to this *National Nuclear Safety Report*, the relationship between ARN and other organizations *remained* the same. In this context, ARN has continued participating mainly in forums and specific Regulatory Bodies' meetings like the Ibero-American Forum of Radiological and Nuclear Regulatory Agencies, CANDU Seniors Regulators, etc.

Its activity has been particularly intense in the Ibero-American Forum, and participated, since 2016 until now, in the following meetings:

EVENT	***PLACE***	***DATE***
Plenary *Technical Committee*	*Montevideo, Uruguay*	*June, 2016*
Technical Committee	*Santiago de Chile, Chile*	*January, 2017*
Plenary *Technical Committee*	*Buenos Aires, Argentina*	*July, 2017*
Technical Committee	*Asunción, Paraguay*	*November, 2017*
Plenary *Technical Committee*	*Brasilia, Brazil*	*July, 2018*
Technical Committee	*Bogotá, Colombia*	*December, 2018*

A number of agreements with domestic and foreign organizations serve as the framework for their relationship with ARN. A list of the new agreements detailing their respective purpose is shown in Table 3.8.2. for domestic organizations, and in Table 3.8.3. for foreign organizations.

Table 3.8.2. – Agreements with domestic organizations

AGREEMENT COUNTERPARTS	AGREEMENT OBJECTIVES
Cooperation Agreement between the Ministry of Health and ARN. August, 2017.	*Cooperation on areas of mutual interest, training, radiosanitary medicine, medical response to emergencies and research.*
Collaboration and Cooperation Agreement between the University of Buenos Aires, College of Architecture, Design and Urbanism and ARN. September, 2017.	*For academic, scientific and technical cooperation.*
Specific Agreement between Airports Argentina 2000 and ARN. September, 2017.	*Assignment by AA2000 to ARN of a space at the Salta International Airport for the construction, installation and operation of the radionuclide monitoring station RN02, part of the CTBT.*
Cooperation Agreement between the Torcuato Di Tella University and ARN. July, 2018.	*Education and training including post-graduate courses of the University.*
Cooperation Agreement between the Airport Security Police and ARN. July, 2018.	*Training courses for joint and coordinated execution of technical projects in the areas of their competences.*
Memorandum of understanding between Nucleoeléctrica Argentina S.A. and ARN. November, 2018.	*Base for licensing a HPR1000 (Hualong I) Nuclear Power Plant.*
Agreement on Electronic Information Exchange between the Nuclear Regulatory Authority (ARN) and the National Social Security Administration (ANSeS). February, 2019.	*Establish reciprocal collaboration between both parties, in order to create and maintain the Register of Workers and / or Beneficiaries of the Pension System for Scientific and Technological Researchers.*

Table 3.8.3. – Agreements with foreign organizations

AGREEMENT COUNTERPARTS	AGREEMENT OBJECTIVES
Arrangement between the Autoridad Regulatoria Nuclear (ARN) of the Republic of Argentina and the National Nuclear Safety Administration (NNSA) of the People's Republic of China for technical cooperation and exchange of information in nuclear regulatory matters. September, 2016.	*To cooperate in matters of mutual interest concerning regulatory aspects in the uses of nuclear energy.*
Modification 1 to the Practical Arrangements between the International Atomic Energy Agency and Autoridad Regulatoria Nuclear, Argentina on cooperation in the area of radiation safety and monitoring. September, 2016.	*To broaden the scope of the Practical Arrangements to include secondary cancers.*
Arrangement between the USNRC and ARN for the Exchange of Technical Information and Cooperation in Nuclear Regulatory and Safety Matters. September, 2018	*Renewal of 2013 Arrangement.* *To establish a framework for the Parties to cooperate in matters of mutual interest concerning regulatory aspects in the uses of nuclear energy.*

Most of the agreements referred to in this and previous reports include clauses for their automatic renewal.

Argentina attaches great importance to its participation in the technical committees and in the commission, working within IAEA's Secretariat-established process for the preparation and review of safety standards *and guides*. These are:

- Commission on Safety Standards (CSS).
- Nuclear Safety Standards Committee (NUSSC).
- Radiation Safety Standards Committee (RASSC).
- Transport Safety Standards Committee (TRANSSC).
- Waste Management Safety Standards Committee (WASSC).
- *Emergency Preparedness and Response Standards Committee (EPReSC).*
- Nuclear Security Guides Committee (NSGC).

Experts from Argentina are also present in the *International Nuclear Safety Advisory Group* (INSAG) and the Standing Advisory Group on Safeguards Implementation (SAGSI).

ARN personnel are also frequently called by IAEA as experts for technical assistance missions to other countries, preparing safety-related publications and providing training for foreign trainees.

As was mentioned in section 3.8.3.1.5., every year ARN offers Post Graduate Courses on Radiological Protection and Nuclear Safety, which started in 1980.

3.8.8. COMPLIANCE WITH THE OBLIGATIONS IMPOSED BY THE CONVENTION

ARN, entrusted with the implementation of the legislative and regulatory control, has been designated in the country. This institution is provided with enough authority, technical and legal competence, human and financial resources to carry out its assigned responsibilities with independence from any other entity concerned with the promotion or utilisation of nuclear energy.

Therefore, the country complies with the obligations imposed in Article 8 of the Convention on Nuclear Safety.

3.9. ARTICLE 9: RESPONSIBILITY OF THE LICENSEE HOLDER

Each Contracting Party shall ensure that prime responsibility for the safety of a nuclear installation rests with the holder of the relevant license and shall take the appropriate steps to ensure that each such license holder meets its responsibility.

3.9.1. BACKGROUND

At the beginning of nuclear activity in Argentina, the small installations had neither the complexity nor the characteristics that could make accidents with significant radiological consequences conceivable in the public.

The responsibility for radiological and nuclear safety of such installations was assigned to one person, generally the installation manager, who by himself or with the help of his personnel or contracting third party services, carried out all the safety related tasks. The Regulatory Body required that such a person should be duly qualified, providing him with the corresponding individual license and authorization, which certifies his qualification.

Moreover, when the design, construction and pre-operational tests of an installation demonstrated to be satisfactory for the Regulatory Body, the corresponding operating license or authorization was granted.

Though these concepts are still essentially valid for smaller installations (low risk installations), several improvements have been introduced to the regulatory system as time went by.

Thus, when the operational characteristics of installations make it advisable, the Regulatory Body requires that those persons holding certain positions in the operation chart must receive specialized training and have their own individual license.

On the other hand, for the case of NPPs, the Regulatory Body considers that it is not sufficient to have enough and suitably trained personnel to guarantee its operation as safely as it was originally designed; obviously, technological progress demands a periodic review of design and operation aspects in such kind of installations and, if corresponds, the introduction of the “state of the art” of safety modifications deemed advisable. These considerations led to the creation of the figure of the Licensee.

NA-SA is the Licensee of CNA I, CNA II and Embalse NPPs, while CNEA is the Licensee of CAREM reactor.

3.9.2. LICENSEE AND PRIMARY RESPONSIBLE

The Regulatory Body requires that each NPP is sustained by an organization capable of providing its personnel with the necessary support for the fulfilment of those tasks related to radiological and nuclear safety, such as the revision of operation procedures, maintenance of safety systems, technical modifications of the plant, etc. in order to increase safety.

The Regulatory Standards AR 0.0.1. and AR 10.1.1. establish the Licensees responsibilities, being the follows the significant ones:

- The Licensee must do whatever is reasonable and compatible with its possibilities regarding safety, fulfilling at least with standards and requirements issued by the Regulatory Body. Such responsibility extends to the stages of design, construction, commissioning, operation and decommissioning of the NPPs.
- The fulfilment of Regulatory Standards, procedures and other mandatory documentation is a necessary but not sufficient condition concerning the Licensee’s responsibility, which must do whatever reasonable and compatible with its possibilities regarding safety. Besides, it must follow the standards and obligations imposed by other competent bodies not related to radiological aspects (e.g. conditions for conventional discharge of chemical effluents).
- The Licensee may support the operation of more than one installation and delegate the execution of the tasks totally or partially, but it must maintain the whole responsibility.

- In each NPP the Licensee must appoint a person of its own body, named Primary Responsible, who will be assigned the direct responsibility for the radiological and nuclear safety of the plant, as well as for the fulfilment of standards, licenses and requirements applicable to it. In the case of a NPP in operation, its plant manager is usually the respective Primary Responsible.
- The Licensee must provide the necessary support to the Primary Responsible in order to allow him to perform his task and responsibility, and must supervise him to verify that he carries out his responsibility satisfactorily.
- The Licensee must submit to the Regulatory Body the technical documents needed to evaluate the safety of the NPP which the operating license is applying for.
- No modification of a NPP related safety system design, operational features or mandatory documentation contained in the operating license, can be initiated without previous Regulatory Body authorization.
- Both the Licensee and the Primary Responsible must facilitate the performance of regulatory inspections and audits, every time the Regulatory Body requires it.
- Every change in the Licensee structure, that could affect its capability of carrying out its responsibilities, must be previously approved by the Regulatory Body.

Moreover, the Licensee must assume the civil responsibility that the Vienna Convention on Civil Responsibility for Nuclear Damages (ratified by Law No. 17,048, 1966) has determined for the licensee. Law No. 24,804 establishes that the Licensee is responsible up to a sum of 80 million US dollars for a nuclear accident, being the State responsible for the remaining responsibility.

Apart from the responsibilities of both the Primary Responsible and the Licensee of a NPP, the Regulatory Body has delimited the responsibilities of workers. In relation with this aspect, the Regulatory Standard AR 10.1.1. establishes that workers are responsible for the fulfilment of those procedures elaborated with the purpose of ensuring their own protection, as well as that of other workers and the public. This subject is consistent with the IAEA recommendations.

3.9.3. COMUNICATION WITH THE PUBLIC

NA-SA carries out a Communication Annual Plan whose actions are intended to keep an open and transparent communication with the stakeholders.

Through different programs, the intention is to install a positive perception of the nuclear power generation. In this context, NA-SA communicates to the public the safe and efficient operation of Atucha Nuclear Power Plant, Units I and II; and the Embalse Nuclear Power Plant Life Extension Project. The more relevant mechanisms applied by NA-SA are the followings:

3.9.3.1. COMMUNICATION CAMPAIGNS

Mass media campaigns are annually developed. The campaigns include radio spots, television, newspapers and social networks.

Besides, NA-SA organizes educational campaigns in the schools that belong to locations near Atucha I NPP, Atucha II NPP and Embalse NPP. This activity consists on providing information regarding nuclear energy to teachers and students. The aim of this is to strengthen links with the areas that are influenced by the facilities as well as disseminate this information among the relatives.

NA-SA participates on the main events of the nuclear industry sector which bring value to the business of the company, through sponsorships, presentations and stands. The company also has presence at the most important events held in the area of influence of the NPPs, with the aim of promoting the development of local areas.

3.9.3.2. RELATIONS WITH THE MEDIA

A strong relationship has been maintained with the media dedicated to the energetic sector, national and local press. In this context, nuclear workshops exclusively for journalists are performed. Public press releases are being emitted to inform on the NPPs´ news and updates.

Radio spots in local radios are periodically disseminated. This action is focused on providing information regarding related themes to the NPPs, management of the company and the nuclear industry in general.

The company also distributes a publication with local newspapers in the area of influence of the NPPs.

3.9.3.3. DIGITAL COMMUNICATION

An active work is performed on digital communication channels belonging to NA-SA. A website is available for the public and a newsletter is sent to subscribers. In addition, NA-SA communication also gathers strength through social networks such as the presence in Facebook, Instagram, Twitter and YouTube.

3.9.3.4. RELATIONS WITH THE COMMUNITY

NA-SA develops a Community Relations Plan, giving a proactive support to enterprises and institutions on areas surrounding the NPPs with the aim of creating conditions for development and welfare of people.

Besides the active Social Corporate Responsibility Program, training presentations regarding the emergency plan in locations influenced by the NPPs are performed. The action is performed as part of the annual training to teachers and students regarding inherent themes on the application of the Emergency Plan. The objective of these presentations is to provide information in a didactic way, strengthening the knowledge related with the individual and collective performances. These presentations take place in a space where the participants can express their concerns and receive the corresponding explanation and answer by specialized personnel.

3.9.3.5. PLANT TOURS

A program of visits to NPPs related to the people is being carried out since many years ago. In *2018* more than *7,000* visitors were received. The general public as well as different institutions like national and international agencies representatives, schools and universities continue participating of this program.

3.9.3.6. COMMUNICATION TO ENVIRONMENTAL AUTHORITIES

NA-SA considers important to include the actions carried out towards the environmental authorities responsible for regulating the instances of preparation and approval of the Environmental Impact Assessment of projects under way, taking into account that in this assessment process, the legislation foresee the potential citizen participation, non-binding, during the public hearing stage.

It should be noted that, according to the legal structure adopted by the Argentine Government in environmental matters, to the Nation corresponds dictate the laws on minimum budgets, and to the Provincial Government to legislate and set specific regulations for the purposes of the aforementioned regulation.

Thus arises, within provincial jurisdiction, the identification of two regulators of environmental impact constituted in application authority for the projects of CNA II (Provincial Agency for Sustainable Development of the Province of Buenos Aires -OPDS-) and CNE Plant Life Extension (Ministry of Environment of the Province of Córdoba).

With regard to the first, in 2015, NA-SA obtained the approval of the Environmental Impact Assessment by the authorities of OPDS.

Regarding the CNE Life Extension Project, and in the framework established by the Provincial Environmental Act. No. 7,343 and its Decree Regulation No. 2,131/00, NA-SA has undertaken with the Ministry of Environment of the Córdoba Province a series of activities (dissemination events, site visits and technical meetings) in order to establish a methodology that facilitates the definition and the agreement of the terms of reference for the corresponding Environmental Impact Study. In 2016, *the Ministry of Environment of the Province of Córdoba approved the Environmental Impact Assessment presented by NA-SA that same year.*

3.9.3.7. COMMUNICATIONS TO OTHERS GOVERNMENT ORGANIZATIONS

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 robustness of the communication system outside the site during a severe accident was checking and the construction of local emergency control centres has been assessed. (see section 3.16.10.2.).

3.9.4. REGULATORY CONTROL ON THE FULFILMENT OF THE LICENSEE RESPONSIBILITIES

Since 1958, the Regulatory Body controls the fulfilment of regulatory standards, licenses and authorizations granted. In order to verify if the Licensees fulfil their responsibilities, the Regulatory Body carries out different types of controls, detailed as follows:

- The Regulatory Body has constantly updated information of the operation organization chart of the installations. The operating license sets that any modification to the organization chart must be reported to the Regulatory Body thirty days before the date of execution. Besides, these modifications are usually known in advance by the Regulatory Body either through the routine meetings held with the Licensee or via the resident inspector's report.
- The Regulatory Standard AR 0.11.1. sets the requisites to be fulfilled by the NPP personnel in order to obtain the corresponding Individual Licenses and Specific Authorizations, according to Section 3.7.3.2.3.
- The procedure of issuing Individual Licenses and Specific Authorizations allows the Regulatory Body to control the aptitude of those persons who must assume responsibilities concerning safety. This aptitude is again evaluated when the Specific Authorization is renewed.
- The Individual License might be revoked by the Regulatory Body in the event that falsified or incomplete background information of the licensee was submitted during the license application process. In the same way, the Specific Authorization may be suspended or cancelled by the Regulatory Body if the licensee fails in fulfilling any condition imposed for such authorization during the execution of authorized tasks.
- Besides, the Regulatory Body carries out a permanent verification that the Primary Responsible fulfils the responsibilities related to safety, and particularly the requirements emerging from the applicable regulatory standards, the operating license conditions and any other conditions related to radiological safety. This is done through regulatory audits and inspections carried out by resident inspectors and analysts of the Regulatory Body.
- The Regulatory Body also verifies that the Licensee fulfils its main responsibilities related to safety through inspections, audits and assessments.
- Moreover, the Regulatory Body performs a permanent follow-up of the Technical Revision Committee and the Internal Safety Advisory Committee minutes (see article 3.10.).
- The Regulatory Body has also issued a sanctions regime for the case of non-fulfilment of any of the regulatory requisites, according to what is expressed in Section 3.7.4.

In summary, ARN in order to control the compliance with safety responsibilities of the Licensee undertakes measures and actions through a combination of assessments, audits, inspections, and enforcements activities. Such activities are performed within the regulatory and legislative frameworks described in article 3.7.

3.9.5. COMPLIANCE WITH THE OBLIGATIONS IMPOSED BY THE CONVENTION

The preceding considerations enable to conclude that the Regulatory Body has taken the appropriate steps to ensure that prime responsibility for NPP safety rests with the Licensee and to ensure that such Licensee fulfils his responsibilities.

Therefore, the country complies with the obligations imposed in Article 9 of the Convention on Nuclear Safety.

3.10. ARTICLE 10: PRIORITY TO SAFETY

Each Contracting Party shall take the appropriate steps to ensure that all organizations engaged in activities directly related to nuclear installations shall establish policies that give due priority to nuclear safety.

3.10.1. GENERAL REGIME

Since the beginning of nuclear activities in Argentina, the State has considered that radiological and nuclear safety of NPPs must be of top priority (also extended to other installations or practices) throughout their stages of design, construction, commissioning, operation and decommissioning.

In principle, this priority to safety is reached through a clear assignment of full responsibility for safety to the licensed organization and the establishment of a national legislative and regulatory framework that is implemented by a strong independent regulatory body provided with adequate authority.

Argentina contributes with international organizations that are part of the Global Nuclear Safety Regime, such as IAEA and ICRP on such matters.

The regulatory body (ARN) establishes and applies a regulatory framework to all nuclear activities developed in Argentina, with the following purposes:

- Protect people and the environment against harmful effects of ionising radiation,
- Keep supervision over radiological and nuclear safety in the nuclear activities developed in the country,
- Make sure that nuclear activities are not developed with non-authorized purposes according to Law No. 24,804, the rules that were consequently dictated, international agreements and non-proliferation policies adopted by the country, and
- Prevent intentional acts that may cause severe radiological consequences or the unauthorized removal of nuclear materials or other materials and equipment of nuclear interest subject to regulation and control.

These purposes are compatible with the global strategy of the regulatory system, which aims particularly at the following basic aspects:

- Regulatory inspections and audits for the verification of the compliance with the respective issued Licenses and Authorizations.
- Independent reviews and assessments in the subjects radiological and nuclear safety, safeguards and physical protection.
- Scientific and technological development in subjects related to radiological and nuclear safety, safeguards and physical protection.
- Personnel training on the above mentioned subjects, for the staff who perform regulatory activities.

As regards the Licensee of the operating NPPs and from the point of view of safety (as shown in the report Policies and Principles of NA-SA) its course of action is such that:

- It complies with pertinent Regulatory Standards and requirements and performs, in addition, all what is reasonable and compatible with its possibilities in favour of radiological and nuclear safety in NPPs, concerning their design, construction, commissioning, operation and decommissioning. To that end, and according to NPPs operation, NA-SA takes the following documents into account:
    - Operating License.
    - Safety Analysis Report.
    - Policies and Principles Manual.
    - Operating Manual.
    - Maintenance Manual.
    - Quality Management System Manual.
    - Radioprotection Manual.

- In-Service Inspection Program.
  - Periodic Test Program.
  - Emergency Plan.
  - Personnel Qualification and Training Program.
- It improves the existing safety practices continuously.
- Ensures that those guides accepted and adopted by the nuclear industry are being fulfilled, when applicable for the case of domestic NPPs.
- It sustains an attitude towards safety based on the organization’s self-evaluation, the feedback of operative experience, technological development and the early prevision of possible degradation of the plants that might affect their safety.
- It continuously carries out training and retraining courses for the plant personnel or for those members of the staff who perform safety related tasks.

Finally, the regulatory system also complies with the concept of Safety Culture, which implicitly results from the before mentioned criteria regarding the ARN and the Licensee.

From the preceding considerations, it should be clearly noticed that the regulatory system (from the point of view of both ARN and the Licensee) guarantees the prioritisation of radiological and nuclear safety in NPPs, in what concerns their design, construction, commissioning, operation and decommissioning stages.

In relation to CAREM 25 prototype reactor, the project management system imposes requirements relating to safety and safety culture that are described below:

Safety

- Safety will be a primary and priority objective within the management system, above all other demands.
- The safety culture must be strong and sustainable so that safety is a primary responsibility or the main focus for all activities.
- The safety shall consist in the implementation of actions to preserve human life, environmental protection and the integrity of the installation, such as:
  - safety considerations during design and construction stages of the nuclear installation,
  - procedures for authorized access control to prevent loss, unauthorized removal, possession, transfer and use of radioactive material personnel,
  - or provisions to mitigate the consequences of accidents and failures, to facilitate measures in dealing with safety breaches that may increase the risks of radiation,
  - measures for the safe management of radioactive sources and materials.

These actions involve all members of the Area Management CAREM with different levels of responsibility and the technical areas of CNEA involved in the CAREM project.

Safety Culture

The management system promotes and supports a strong safety culture:

- Ensuring a common understanding of the key aspects of safety culture within the organization.
- Providing the means by which the organization supports individuals and groups of people in the safe and efficient performance of its tasks, taking into account the interaction between people, technology and organization.
- Strengthening an attitude of awareness and learning at all levels of the organization.
- Providing the means by which organization constantly develop and improve its safety culture.
- Applying the concepts defined in the GS-G-3.5 IAEA standard:
  - Safety is a value clearly recognized.
  - Leadership for safety is clear.
  - The responsibility for safety is clear.
  - Safety is driven by learning.
  - Safety is integrated into all activities.

3.10.2. SPECIFIC ACTIONS

3.10.2.1. SAFETY POLICY

The safety regime described above is fulfilled in every activity related to NPPs. In particular, the priority to safety may be noticed in the Operating License and in the policies and principles manual for each NPP, including the fact that there are operational limits and conditions for any of the NPPs considered.

The ARN establishes in the Operating License that two advisor committees shall exist concerning safety subjects; these committees are:

- Internal Safety Advisory Committee (CIAS).
- Technical Revision Committee (CRT).

The CIAS reports and advices to the Plant Manager, and its members are chosen for their knowledge and experience. This advice is regarding the actions to be followed in, for instance:

- Outages.
- Safety related incidents.
- Plant Design Modifications and Procedures Modifications (safety or safety related systems).
- Abnormal situations.
- Periodic evaluation of the installation performance.
- Periodic evaluation of the training personnel program.
- Emergency plan (including the corresponding exercises), etc.

The advice given by the CIAS consists of analyses, conclusions and recommendations issued as a minute signed by its members.

On the other hand, the CRT, which is independent from the Plant Managers, advises the highest staff level of the Licensee as regards the safe operation of the NPPs, analyses the importance of the failures, abnormal and significant events, evaluates the proposed design modifications that may affect the safety related systems, and in general about the same subjects dealt with by the CIAS.

The CRT is integrated by senior professionals, appointed for their knowledge and experience that constitute the Licensee's technical support. Its conclusions and recommendations are issued as minutes.

Both Committees' minutes are of the highest importance to the ARN, due to the fact that both the Primary Responsible and the Licensee independently produce written evidence of their opinion concerning nuclear and radiological safety related subjects with reference to the particular installation under consideration.

3.10.2.2. SAFETY CULTURE AND ITS DEVELOPMENT

Special attention has been given to safety culture (SC), promoted by the maximum staff level of the Licensee and the Plant Managers to all the personnel. Such promotion is based on diffusion, training and re-training, providing all personnel with the benefits of applying the SC *key aspects* to all activities carried out at NPPs.

The ARN and the Licensee are continuously involved in the compliance with the above mentioned *key aspects*.

Additional attitudes carried out by both institutions that contribute to prioritise safety can be mentioned:

- Evaluation of the SC by the Licensee is included in the program for renewal of personnel Specific Authorizations.
- Evaluation of the SC attitudes during inspections by ARN specialists.
- Evaluation of trends in event reports, corrective action effectiveness and measures implemented to prevent safety problems.
- Evaluation of trends for safety performance indicators.
- Assessment of minor event responses reported by the Licensee to detect organizational weaknesses and inadequacies.
- Increasing use of PSA for plant safety management.
- Improvement in the relationship between ARN and NA-SA.
- Efforts to improve the safety of the NPPs have the highest priority in both ARN and NA-SA.

- The Licensee has been developed a Programme of Consolidation of Safety Culture (PRACS Spanish initials) to reinforce nuclear safety culture. The goal of the PRACS is to create a bridge between the concepts of Nuclear Safety Culture and actual performance in the stations. The programme defines management issues and implementation issues that require improvements. An eight step strategy for each topic is in place. Organization is based on work teams with participation of leaders and individuals with high credibility in the organization. The measurement of the evolution is done through, surveys, self-assessment and *performance reports*.

Additional activities within the SC were carried out to improve the promotion and the evaluation such as:

- Performance self-evaluation. Evolution and growth through the years provides evidence that it is one of the most valuable ways whereby the Regulatory Body can promote Licensee's SC own proficiency, which includes, among other attitudes: professionalism, teamwork, organizational and individual commitment to the SC.
- Regulator – operator relationship. The relationship has been improved using simple approaches such as:
  - Polite and professional attitude in verbal communications.
  - Honest dialogue particularly focused on accomplishing safety objectives more than on strict compliance with rules and promoting good practices for high performance in the plant activities.

3.10.2.3. COMMITMENT TO SAFETY

The commitment to NPPs' safety is made clear in design and operation aspects that give priority to safety concepts over economic profitability.

Examples of this commitment are the compliance with the defence-in-depth principle and the rule that has been observed along the NPPs lifetime according to which it was decided to shut down the plant when a deviation from normal operating conditions occurred.

The compliance with Maintenance Programs, ISI Programs and good operation practices are also part of the commitment.

The commitment to safety is also verified through the assignment of resources for programmed revisions, and for maintaining and improving systems and components related to the plant safety.

3.10.2.4. MANAGEMENT ATTITUDES TOWARDS SAFETY

The most important subjects of the operating NPPs are dealt with as a whole in the periodic Management Meetings, where the importance of safety and the commitment to the achievement of the goals is emphasised.

A follow-up is carried out by the highest level of authority both of the NPPs and the Licensee, concerning the conclusions and recommendations emerging from CIAS and CRT meetings. In these committee meetings, the on-the-job performance of NPPs personnel is evaluated, among other activities, and their improvement is encouraged with the help of the conclusions emerging from the analysis done.

Additionally, periodic meetings at the highest level are carried out between ARN and NA-SA specialists. In such meetings the main safety aspects arisen from regulatory inspections, safety analysis and other assessments are considered. Safety aspects related to NPPs programmed outages, as well as the progress in back-fitting related activities in the installations are also analysed. The conclusions and recommendations are issued as minutes.

3.10.2.5. PERSONNEL MOTIVATION

The motivation of the personnel depends mainly on the attitudes and policies applied by the managers and their adherence with the SC principle.

The attitudes and policies followed by the managing staff concerning personnel thus, tend to reinforce the means of encouraging such adherence, that is:

- Initial training,
- Periodic retraining,
- Exchange of experience (operational, maintenance, etc.),

- Professional prestige (both in their own institution and in the national or international nuclear community), and
- The preservation of the working position and chances for promotion (concerning both technical and pecuniary aspects).

As a result, workers have a generalised conscience about the individual and collective way of acting, concerning radiological and nuclear safety.

3.10.2.6. VOLUNTARY ACTIVITIES AND GOOD PRACTICES RELATED TO SAFETY

Concerning this matter the following practices or activities are worth mentioning:

- Consults and meetings between NA-SA and ARN specialists with the objective to facilitate and improve the compliance with general and specific requirements, evaluating, in addition, the operational situation of the NPPs (as was mentioned before).
- Participation in the IAEA Incident Reporting System, that enables the contribution and return of operational experience, from which some actions may be applied to the domestic NPPs.
- Active participation of the Licensee in international organizations of nuclear operators: the CANDU Owners Group (COG) and the World Association of Nuclear Operators (WANO). Both organizations promote the exchange of operational experience and give technical assistance in response to NPPs requests.
- Implementation of external technical audits, for instance the peer review performed by WANO to the NPPs.
- Interaction with official and non-governmental bodies, with the purpose of analysing emergency preparedness measures, including the role of the ARN and other organizations (i.e. Civil Defence at national level).
- Interaction with the Uruguayan and Brazilian Regulatory Bodies in their character of neighbouring country, with regard to their participation in the practical emergency plan exercises at CNA.

3.10.3. COMPLIANCE WITH THE OBLIGATIONS IMPOSED BY THE CONVENTION

The preceding considerations reflect that in Argentina the appropriate steps have been taken to give priority to nuclear safety, showing compliance with the obligations imposed in Article 10 of the Convention on Nuclear Safety.

3.11. ARTICLE 11: HUMAN AND FINANCIAL RESOURCES OF THE LICENSEE

1. *Each Contracting Party shall take the appropriate steps to ensure that adequate financial resources are available to support the safety of each nuclear installation throughout its life.*
2. *Each Contracting Party shall take the appropriate steps to ensure that sufficient numbers of qualified staff with appropriate education, training and retraining are available for all safety-related activities in each nuclear installation, throughout its life.*

3.11.1. INTRODUCTION

The Licensee of the Argentinean NPPs, NA-SA is a state owned company. The necessary resources to attend the suitable support of the NPPs are incorporated in the NA-SA annual budget which it's approved by the Ministry of Treasury.

These resources cover the acquisition of the necessary supplies and services for the normal development, the planned special revisions as well as the improvements of the NPPs.

Apart from resources that come from electrical generation there are two national laws (Law No. 24,065 and Law No. 24,776) that guarantee all the needs for plant operation, maintenance and, the resources needed to support nuclear safety.

During the period *2016-2018* the industrial electrical tariffs followed the increase of generation costs, while the residential tariffs had a staggered growth that did not cover 100% of the costs of generation.

The difference between the generation costs and tariffs has been absorbed by the National Government using different mechanisms.

The average increase of electrical energy demand in this period was *0.23%* per year, while the installed capacity has an increase of approximately *15.0%.*

This increase came from the installation of combined cycle plants through a joint investment of private capital and the National Government. *Installed generating capacity increased also due to the public tenders carried out by the National Government for thermal and renewable generation projects.*

Referring to nuclear generation, *CNE began its life extension on January 1st, 2016. The works that were carried out will increase the power of the plant by 35 MW and will allow it to continue generating power for another 30 years.*

In Figure 3.11.1. it can be finding the electrical generation from 1975 to *2018,* classified in hydroelectric, thermal, renewable and nuclear.

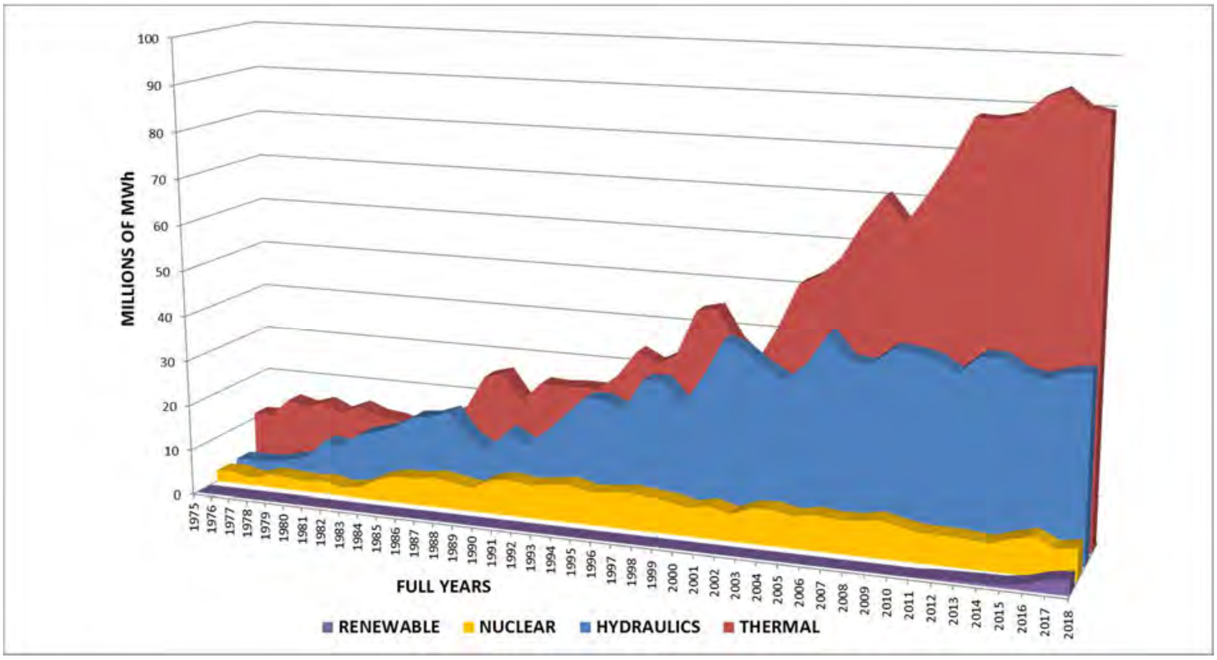


Figure 3.11.1. – Electrical generation in Argentina from 1975 to *2018*

3.11.2. ELECTRICAL GENERATION AND ECONOMICS

3.11.2.1. TOTAL ELECTRICAL GENERATION PERIOD 2016 – 2018

In Figures 3.11.2. to 3.11.4., the total electrical generation is shown by type of source thermal, hydro and nuclear.

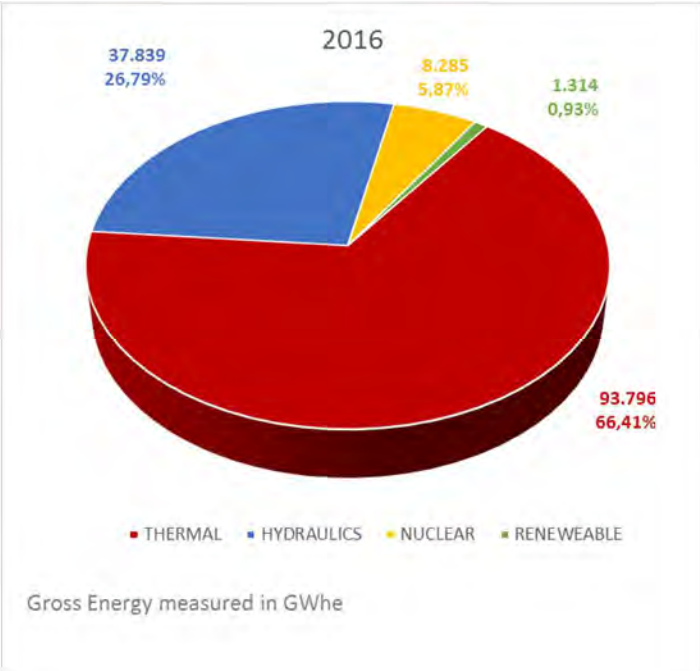


Figure 3.11.2. – Gross energy generated in *2016*

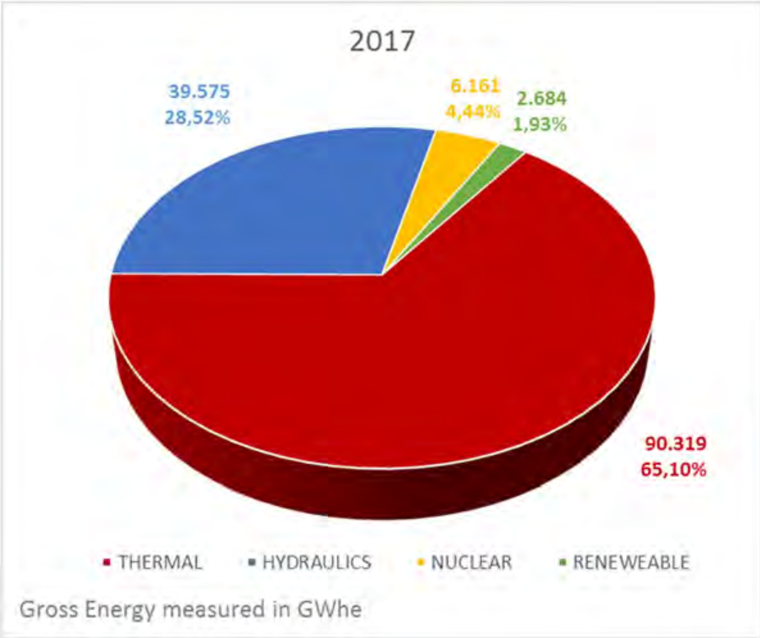


Figure 3.11.3. – Gross energy generated in *2017*

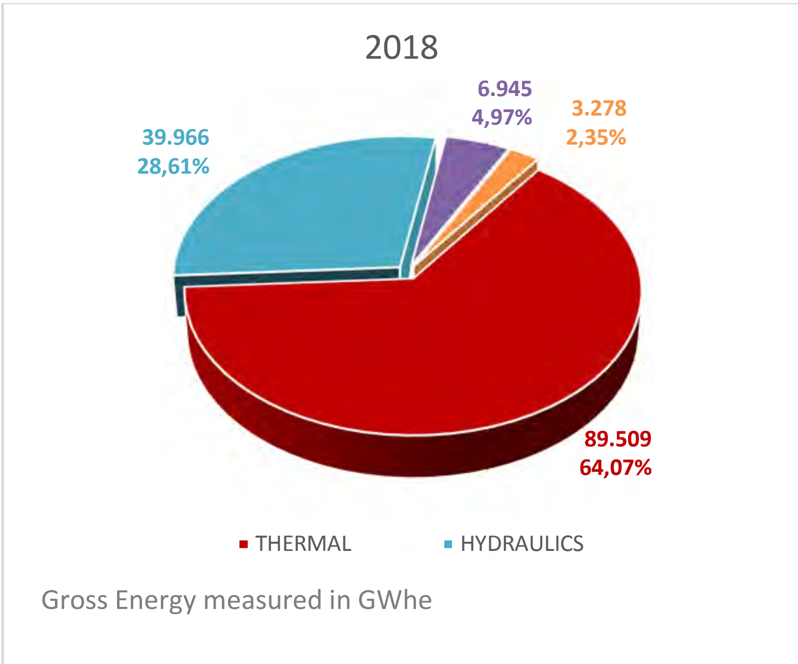


Figure 3.11.4. – Gross energy generated in *2018*

3.11.2.2. NUCLEAR GENERATION 2016, 2017, 2018

Performance of nuclear generation is shown in Table 3.11.1.

Table 3.11.1. – Performance

	2016	2017	2018
Gross Energy (MWh)	*8,284,702*	*6,161,235*	*6,945,308*
Load Factor (%)	*53.74*	*40.08*	*45.18*
Installed Nuclear Power (%)	*5.19*	*4.85*	*4.55*
Generated Nuclear Power (%)	*5.87*	*4.44*	*4.97*

3.11.2.3. HUMAN RESOURCES

After several years without incorporation / renewal of personnel, considering the average age of employees and staff close to retirement, NA-SA requested authorization to develop a 5-year program in order to incorporate new professionals. This program covered the period 2002-2007. Continuing with the implementation of the program, the process of hiring and training of junior professionals, already mentioned *in the previous 2013-2015 report, was stabilized. During the 2016-2018 period, the NA-SA staff has remained stable, tending to decrease due to retired personnel as shown in Table 3.11.2.*

Table 3.11.2. – Personnel *2016, 2017 and 2018* by work area and specific knowledge

	Year	CNA I - II	CNE + PEV	Project IVCN	Main Branch	TOTAL
Professionals	2016	401	364	209	214	**1188**
	2017	385	341	209	207	**1142**
	2018	373	324	160	207	**1064**
Technicians	2016	625	396	173	38	**1232**
	2017	609	379	174	31	**1193**
	2018	604	328	75	29	**1036**
Administrative personnel	2016	445	326	252	118	**1141**
	2017	428	315	252	105	**1100**
	2018	416	270	120	90	**896**
Total	**2016**	**1471**	**1086**	**634**	**370**	**3561**
	2017	**1422**	**1035**	**635**	**343**	**3435**
	2018	**1393**	**922**	**355**	**326**	**2996**

Taking into account the projects, NA-SA completed the Life Extension of the CNE and the plant is in commissioning face. The reactor has reached full power and is performing the full power tests. In addition, it continues with the Feasibility Assessment for a Fourth NPP, the pace of incorporation of new personnel is stable. It should be emphasized that after completing and commissioning the CNE, the staff is ready to work in positions that require special authorization from the ARN, who participated in a strict training program.

Having all these requirements in mind, several training courses have been organized through specialized centres, such as the Instituto Dan Beninson (Universidad Nacional de San Martín) which provides training in nuclear reactors and fuel life cycle.

NA-SA signed agreements with universities such as the National University of Rosario and University Torcuato Di Tella, to train professionals and technicians, not only in technical aspects, but also in management skills.

Furthermore, NA-SA has an agreement with Sabato Institute in which people from materials engineers studies make their thesis in the company. This situation has mutual benefits, students can obtain their degree and NA-SA can keep the results of the research.

During 2019 NA-SA has been contacting people from Consejo Nacional de Investigaciones Científicas y Técnicas (CONICET) to sign an agreement between both institutions. Nowadays, NA-SA is analysing the formalities to elaborate the document.

In the particular case of CNE, the following processes were developed in order to have trained personnel to carry out the Start-up and second cycle of operation of the plant:

- *Specialized courses given by the University Centre for Nuclear Technology (National University of Córdoba): initial training in nuclear technology, aimed at incoming professionals and technicians; complementary training aimed at personnel who will occupy positions requiring specific authorization from ARN.*
- *Updating of the specific training of CNE technicians and professionals on the systems modified or incorporated by the CNE Life Extension: training given by the specialists assigned to the modification of the systems, aimed at 245 people belonging to Operations, Maintenance, Chemistry and Processes, Engineering, Emergency Preparedness, Programming and Planning, Training.*
- *Specific training in Full Scope Simulator updated to the new CNE configuration, aimed at the 33 people occupying positions in the Main Control Room, given by simulator instructors trained by TECNATOM S.A. (Spain).*
- *Training in business management given by the Blas Pascal University (Córdoba), aimed at all personnel occupying positions of Head of the CNE.*
- *Specialized support for the Life Extension and Start-up of the CNE, with foreign personnel coming from: CANDU ENERGY, ANSALDO NUCLEARE, ABB, SCHNEIDER, L3MAPS, AREVA, ANDRITZ AG, CATERPILLAR, MAMOET, PCI and national organizations such as INVAP and UNLP (National University of La Plata), among others.*

Besides, several negotiations were carried out with CNEA, the Instituto Balseiro, as well as international companies, in order to obtain support for the training, either in preparing the materials, or in providing the simulation tools.

In the recent past the main source of loss of human resources has been the personnel retirement. Very few resources were lost due to man power transfer to other industries. To compensate for this fact NA-SA has been incorporating personnel which were trained by the experienced staff next to the retirement.

3.11.2.4. CURRENT EXPENSES 2016, 2017, 2018

Figure 3.11.5 shows the evolution of Operating and Maintenance costs

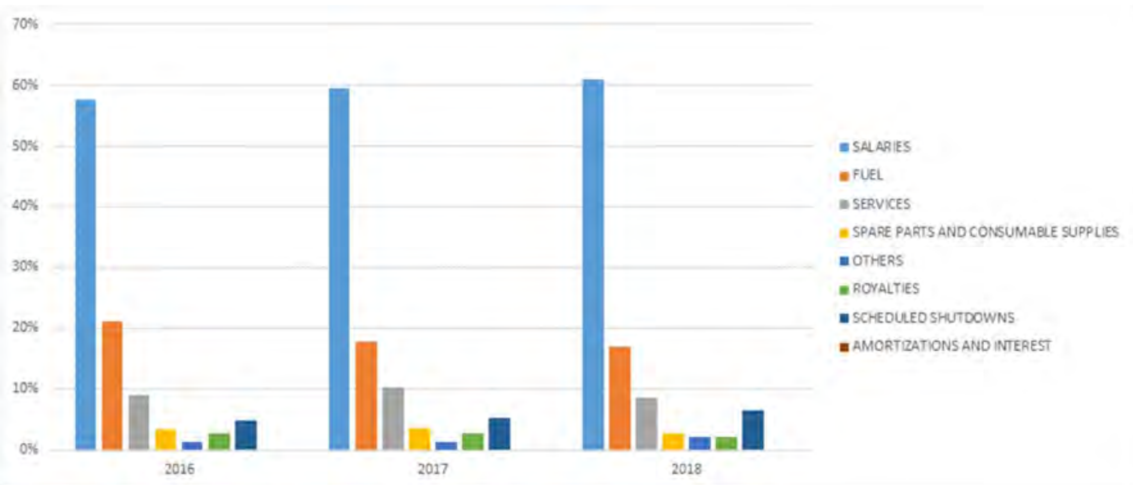


Figure 3.11.5. – Evolution of O&M costs, period *2016 – 2018*

3.11.3. COMPLIANCE WITH THE OBLIGATIONS IMPOSED BY THE CONVENTION

The nuclear electric generation maintained its quality level both in the safety and availability of the power stations, meeting to all the regulatory requirements.

The preceding considerations show that the Licensee has taken the appropriate steps to ensure that adequate financial resources are available to support the safety of each nuclear installation, and a sufficient number of qualified staff with appropriate education, training and retraining are available for all safety-related activities in each nuclear installation, throughout its life, showing compliance with the obligations imposed in Article 11 of the Convention on Nuclear Safety

3.12. ARTICLE 12: HUMAN FACTORS

Each Contracting Party shall take the appropriate steps to ensure that the capabilities and limitations of human performance are taken into account throughout the life of a nuclear installation.

3.12.1. SYSTEM TO DETECT, CORRECT AND PREVENT HUMAN ERRORS

The detection, prevention and correction of human errors are carried out through two processes: the analysis of incidents and the global and systematic assessment of the installation safety.

The observation of tasks by area leaders, the rigorous reading of the weekly reinforcement of human performance, the field training on the use of different error prevention techniques by the Human Factors area, make up a proactive process aimed at reinforcement of expectations and improvement of performance in the field.

In the first case, the process acts on abnormal or unexpected events that happen in the installations (operating experience, OPEX). Such events are unique opportunities to detect and as consequence to correct human errors, identifying the deficiencies regarding organization, people, materials and practices. In this case the key elements are the quality of the report on the occurred events, the rigour in the investigation of their root causes and the corrective actions carried out.

Because of the importance a good root cause analysis has for the improvement process, specific requirements have been established for analysts:

The analysts who carry out investigation of root causes are experts in the event analysis, by using among other issues, knowledge of systems / components; follow up of event sequences and group brainstorming. The most applied methods are those related to HPES (Human Performance Evaluation / Enhancement System) and WANO root cause methodology.

The analyst is assisted by the Human Factors area to apply the root cause analysis tools and for human performance to be analyzed in said event. Finally, this analysis is evaluated to ensure that it meets the quality standards intended by the organization.

In the second case, the PSA technique is used, part of which consists in the identification of human actions and the evaluation of their relative importance on the installation's safety. The errors can be classified in pre-accidental (errors occurred during periodic tests or maintenance tasks) and post-accidental (errors occurred during abnormal or unexpected events).

These pre-accidental and post-accidental errors are analysed in the same way as the behaviour of SSCs, but using human reliability analysis techniques. Those evaluations are part of the PSA and their results enable the definition of those areas requiring improvements on both the operation procedures and the man-machine interface, as well as the identification of cases in which the operators' training and retraining should be intensified.

Lessons learned and corrective actions are followed-up by assessments, inspections and regulatory audits. Moreover, considering the experience gathered during the CNA I, CNA II and CNE PSAs, the periodic training of CNE personnel at Gentilly-2 simulator in Canada, CNA I personnel at Angra simulator in Brazil, CNA II personnel in the new own simulator, as well as the human reliability analyses carried out for CNA I and CNE PSAs in shutdown state, it may be concluded that important steps have been taken to ensure that the capabilities and limitations of human performance are taken into account.

Additionally, PSAs carried out for CNA I, CNA II and CNE show that human corrective actions were considered necessary in order to ensure that the capabilities and limitations of human performance were taken into account in the procedures for normal and abnormal operation.

Furthermore, Regulatory Standards AR 3.2.1. and AR 3.4.1. establish the information the operator should count with in order to take safety related decisions, the prohibition of interventions during the period immediately after the occurrence of accident initiating events and the characteristics of the man – machine interaction related to the design of the reactor instrumentation and protection systems.

Another aspect to take into account in the prevention of human errors is related to the measures taken by the Licensee regarding the contractors in order to ensure their adequate competence and safety culture. Towards this objective, the technical assistance services to the NPPs are given by competent companies whose personnel has qualifications, knowledge and experience about domestic NPPs.

Also the new personnel at any level are trained with the necessary knowledge before they enter the nuclear area.

3.12.1.1. HUMAN CORRECTIVE ACTIONS IN CNA I

Many human action improvements were described in previous reports. The back-fitting program included a large number of design changes and procedure modifications. After the back-fitting implementation, the evaluation of the new main contributors to the core damage frequency permitted to consider the most safety related human actions. In this sense, the reliability of such actions was carried out modifying procedures and increasing training efforts. Additionally, new systems were included and consequently new procedures were carried out improving the overall plant safety. The size of the back-fitting program implied the review of all plant procedures and the critical human actions that, combined with the PSA results, allowed improving its reliability. Main changes related to procedures were following:

- Emergency procedure modifications and update such as “loss of feed water system”, “loss of house-pumps” and “loss of off-site power”.
- An additional test was included regarding the second heat sink (emergency feed water system).
- Procedure improvements within the Surveillance Program.

The Licensee has established a program to evaluate the incidence of human factors in the safety performance of the NPPs. The program mainly comprises:

- Evaluation of low safety significant events, near misses and operational events to find the human factor related causes, in a systematic way. These evaluations are used as a learning method to prevent reoccurrence of safety significant events.
- Lessons learned from such events are used as feedback to prevent recurrence. To reach this objective, plant personnel receive specific retraining regarding the causes and consequences of each event.
- Identification of precursors to operational events. Some human factors related to low safety significant events and “near misses” can be correlated to organizational deficiencies, therefore these events may be used as leading indicators in anticipating and preventing declining performance.
- Detection of Organizational (human-related) deficiencies shows how safety must be managed to help avoiding mistakes and preventing incidents.
- *The coding of causes and contributors of all the events that occurred, the analysis of international codes and additional codes associated with human performance to identify corrective actions.*
- *In trend analysis based on task observations of leaders in the field to identify actions to improve human performance in areas of articulation or cross-cutting to the organization.*
- *Work reports to establish common issues of human performance in isolated situations.*
- *Integration of isolated processes that provide information on human performance to drive from integrated and complete analyses*.

To evaluate incidence of human factors in low-level events, near misses and operational events, the most commonly used methods were adjusted to the different specific situations. The systematization of the evaluation process involves the use of international applicable methodologies to evaluate human performance in CNA.

Combination of Human Performance Enhancement System (HPES) and Human Performance Investigation Program (HPIP, similar to the first one but used by regulators) is still considered sufficient to detect both human factors and organizational deficiencies as “root causes” and “contributing causes” of analysed events. Using these methodologies and their associated techniques, it is also possible to find the adequate corrective actions to be taken.

As it was reported in Fifth National Report, the implementation of a Human Performance Improving Program (HPIP) has been continued, aiming to promote certain behaviours at all levels of the organization to support a plant safe and reliable operation. In order to achieve this objective, the Management has revised General Behavior Expectations for the Site, developing the specific expectations of each Sub-Management and Department, providing a structured framework of expected behavior. They are communicated through presentations by the Site Manager, Refresher training courses by senior officers of the Site, *weekly reinforcement of human performance, reinforcement of expectations through the digital and printed billboard system available throughout the CNA site.*

The HPIP comprises the definition and spreading of a set of Human Error Reduction Tools, Task Observation Program, and the exploitation of feedback information from other already implemented programs, such as the Operational Experience Program. To sum up, the program encompasses three main areas:

1. Human Error Reduction Tools

The following Event-Free Tools were defined as part of the program, and the personnel trained in their application:

- Pre-job briefing and Post-job debriefing.
- Procedure use and adherence.
- *Effective Communication (3-way communication - phonetic alphabet).*
- Checking techniques:
  - Self-checking.
  - Peer-checking.
- Conservative decision making.
- Questioning Attitude.
- Independent Verification.

The following activities are periodically performed:

- Self-checking evaluation on the application of the Human Error Reduction Tools during programmed outages.
- Training and retraining of all personnel in the different techniques and methods of application in the field of action in order to minimize the presence of human error.
- Annual Agenda of weekly reinforcement of different techniques, incorporating new concepts in relation to the techniques used, spreading successful operating experience as a result of adherence to technology, as well as Plant events resulting from non-adherence. This information is weekly spread and is used at the beginning of each Plant meeting to regularly and systematically reinforce expectations of human behaviour and the importance of applying the techniques of human error prevention.
- Half-yearly trend analysis of events whose direct cause is human factor, specifying the type of associated behaviour, identifying adverse trends and taking action to reverse such trend and monitoring.
- Monitoring the application and use of the techniques in the field through task observation (item 3), reinforcement and coaching in the field by process leaders, trend analysis by area, identifying needs and taking action.
- Internal self-assessments to monitor the performance of operating processes associated to the deviations identified.
- Exchange of experience between NA-SA Managements to standardize processes, adding successful practices of other Managements (PRACS).

2. Operational Experience Application

Internal Operational Experience is used within the HPIP to recognize human conduct problems and behaviour enhancements. A half-yearly trend analysis is carried out to assess those events whose direct causes and/or root causes or causal factors are due to human error. The process and method of calculation for identifying trends, has been formalized through the development of procedures and internal instructions of the Site. *The procedures describe how to standardize the coding of causes and contributors of all the events that occurred, taking not only the international codes on human performance, but associated additional codes to identify corrective actions.*

The methods used in the industry to compare results of this Site with international industry trends have been taken as a reference.

Taking into account that one application of operational experience is its discussion during pre-job briefings; its insertion into work packages is considered part of this program and will be rigorously applied in work management process of the Site.

3. Task Observation

A Program for Task Observation with the following objectives is being implemented:

- Annual retraining of observers, incorporating new practices to implement coaching focused on improving teams.
- General training to all management lines on issues related to team leadership.
- Task Observations to observers to ensure and reinforce the correct use of the tool.
- Exchange of experience between NA-SA Managements to standardize processes and add practices of other successful Managements (PRACS).
- Strengthen the plant's standard and expectations.
- To ease communication between chief-staff and job executors.

Bimonthly analyses with each Sub-Management and half-yearly with Management are conducted to monitor compliance with the program expectations; the necessary corrective actions are formalized in the database of the Site for further follow up.

Based on the operating experience of the last years, the Task Observation procedure has been updated. Managers and supervisors were trained in the new procedure.

The expectation regarding the number of observations required for each manager and supervisor was clearly defined by the Station Manager and formalized as internal instructions of the sector making them available to be consulted by the complete Plant Personnel throughout the plant database which is part of Site documentation.

In 2019 the SOP was implemented: Strategic Observation Program that contains a subgroup of the total number of task observations made by each leader in the field and responds to a common focus defined at the beginning of the year by the site manager, in this line, all the observers contribute their findings in order to verify the progress of this line in the different areas. The program of strategic observations is one of the actions aimed at ensuring the quality and usefulness of the interventions of the leaders in the field.

3.12.1.2. HUMAN CORRECTIVE ACTIONS IN CNA II

Regarding Human Factors, from 2013 to date, Human Performance has developed as a unified program in both units. All the programs described in Section 3.12.1.1. belong to Atucha NPP Site and are applicable to both units.

3.12.1.3. HUMAN CORRECTIVE ACTIONS IN CNE

It is essential to work on aspects related to the performance of employees, at any level, as a necessity to operate nuclear power stations with the best levels of safety and productivity.

With the application of Human Performance Program, CNE aims to:

- Reduce the frequency of events by anticipation, prevention and detection of active error in the work site. This involves significant human interactions with nuclear safety, security, environmental security and generating capacity.
- Minimize the severity of events by identification and elimination of latent weaknesses which impede the effectiveness of the defense against human error and its consequences.

CNE established different processes with the objective of improving human performance:

- Rounds specified in the Management Manual.
- Operative Experience.
- Work Management.

- Task Observations.
- Self-evaluation process.
- Condition monitoring of materials of equipment, components, and structures process.
- Foreign Material Exclusion process.
- Making of Operational Decision.

The program's effectiveness feedback is obtained from the conclusions arisen for the different processes established at CNE. Information from self-evaluations, task observations, internal, external, and audit operational experience is used to formulate the needs for corrective actions.

A better understanding of the results of the Human Performance Program can be learned through the information gathered from the management indicators.

CNE developed the following procedures:

- Human Performance Fundamentals Procedure.
- Event – Free Tools Procedure.
- Pre-job briefing and post-job debriefing Procedure.
- Task Observation Procedure.
- Self-assessments Program.

CNE has been working in the training of its staff on subjects related to human factors. One of the approaches taken has been through the attendance to seminars, some of which are listed below:

- Team work.
- Human Performance Improvement.
- Conservative decision making.
- Communication.
- Safety Culture.
- Self-checking.
- Leadership.
- Diagnosis.
- Conflicts resolution.
- Motivation.

To evaluate incidence of human factors in low-level events, near misses and operational events, the most commonly used methods were adjusted to the different specific situations. The systematization of the evaluation process involves the use of international applicable methodologies to evaluate human performance in CNE.

At CNE, the Human Performance Program consists of the following:

- Use of Error Prevention Techniques.
- Management in the field.
- Self-Assessments.

3.12.1.3.1. Error Prevention Techniques

In the last period, some activities continue being carried out to improve human performance at CNE:

- In 2013, the Three-way communication tool was modified to effective communication.
- In 2014, a new Error Prevention Technique joined in CNE, "Taking of conservative decisions".
- In 2015-*2016*, the Training Department gave a retraining called "Back to Basics". It was addressed to CNE staff in order to strengthen the use of Error Prevention Techniques (Pre-job briefing and Post-job debriefing, Self-checking, Peer-checking and Independent Checking, Use and Adherence to the Procedures, Use Three-way Communication, Questioning Attitude, Making Conservative Decision) to generate a tangible link between staff and daily activities, as well as also to impart a healthy concern towards the human fallibility and vulnerability. The program included 12 hours of retraining divided into a theoretical block and a practical one.

- Human Performance Champions were appointed in each sector of CNE in order to promote the use of Error Prevention Techniques.
- Every week, Training Department sends the "Weekly Safety Culture Strengthening Message" aiming to install the use of an Error Prevention Technique. Also, in each meeting or training in plant it must make comments on the tool of the week. *In addition, NA-SA have an Annual Agenda of weekly reinforcement of different techniques, incorporating new concepts in relation to the techniques used and expectations of management in relation to the safety culture.*
- *In 2016/2017, personnel hired for the life extension project were trained in the use of Human Performance tools through the Hygiene and Safety Committee.*
- *Exchange of experience between NA-SA Managements to standardize processes and add practices of other successful Managements (PRACS).*

3.12.1.3.2. Management in the field

The Management in the Field Program was developed with the aim of ensuring the management of quality through the observation of tasks in the field.

The activities carried out were:

- Defining expectations for Task Observations according to position in the Organization chart.
- Training in Task Observations to CNE partners in the framework of the “Back to Basics” activity.
- Retraining through a Technical Support Mission (TSM) oriented Coaching by WANO.
- Quality Analysis of the Task Observations.
- Observation cards were set, focused on different sectors such as Operation, Maintenance and Training, among others.
- *Monthly Analysis with each Sub-Management are conducted to monitor compliance with the program expectations.*
- *Exchange of experience between NA-SA Managements to standardize processes, adding successful practices of other Managements (PRACS).*

3.12.1.3.3. Self-Assessment

Self-Assessments Program is carried out annually. It is a structured, objective and transparent process where the staff evaluates its own effectiveness in comparison with the previously established one.

In 2018, a methodology was developed to carry out a Corporate Self-Assessment on the Culture of Corporate Security.

Exchange of experience between NA-SA Managements to standardize processes and add practices of other successful Managements (PRACS).

3.12.2. MANAGEMENT AND INSTITUTIONAL ASPECTS

The proper policies and management of the Licensee are the basic support to obtain the expected results regarding the anticipation of undesirable events that may happen.

Once such events have happened, the NPP´s Primary Responsible determines the responsibility degree, if any, of persons who may have incurred in errors and applies the corrective measures and, if it corresponds, the pertinent sanctions.

On the other hand, having analysed the event, the ARN issues requirements and, if it is deemed necessary, applies the corresponding sanctions to the involved personnel, the Primary Responsible and the Licensee.

During the safety inspection and evaluation process of the NPP, the ARN pays special attention to find early signals and trends such as:

- Weaknesses in the safety policies.
- Weaknesses in accident analyses.
- Procedure violation.

- Operator errors.
- Deficient training.
- Deficiencies in the use of operational experience.
- Weaknesses in emergency planning.

3.12.3. HUMAN RELIABILITY ANALYSIS (HRA)

The aim of the human reliability assessment is to improve the NPP´s global safety, identifying deficiencies in the operator actions and providing whatever is needed to analyse and perform possible corrective actions.

The HRA of manual operations outside the main control room are included in the PSA. Mostly, PSA studies implemented in Argentinian NPPs include HRA based on identification of the human actions specified in the *documents:*

- *IAEA-Safety Series 50-P-10 "Human Reliability Analysis in Probabilistic Safety Assessment for Nuclear Power Plants".*
- *IAEA Specific Safety Guide SSG-3 "Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants".*
- *EPRI NP-3583 (NUS Corporation) "Systematic Human Action Reliability Procedure (SHARP)".*
- *NUREG CR/4772 "Accident Sequence Evaluation Program Human Reliability Analysis Procedure".*
- *NUREG CR/1278 "Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications".*

In this sense, the typical Human Actions (HA) considered are:

- Category A (pre-initiators) including maintenance, test, calibration, realignment and restoration during normal operation.
- Category C (post-initiators). These human actions were mainly incorporated as "human headings" in the event trees. They could be operator diagnosis, operator actions and recovery actions in some specifics cases.

In this context, most manual operations outside the control room are included in Category A. On the contrary, most of Category C is related to operator actions carried out inside the control room. Only some of them may require some specific support tasks outside the control room. Besides HRA is used to identify deficiencies in the operator actions, including manual operations outside the control room, and to provide all that is required to analyse and perform possible corrective actions.

The PSA of CNA I showed by the application of human reliability analysis, that it was necessary to carry out modifications to the installation enabling the operator to take more reliable protective measures, to make improvements regarding abnormal operating procedures and re-training the operating personnel on certain analysed accidental sequences, where human actions play an important safety related role.

In the PSA of Atucha II NPP the same considerations as in the PSA of CNA I were applied and their results were used to develop procedures and re-train operators.

The data used in human reliability models depend explicitly on the applied model and are obtained from operational experience, generic data and practices in simulators. Specifically, the human reliability analysis carried out for CNA I and CNE PSAs were based on generic data for the human error failure rate, from factors, recovery and uncertainty factors. CNA I and CNE OPEX provide task execution times, performance frequency for components and equipment and equipment recovery times.

3.12.4. PROGRESS IN HUMAN FACTORS

The progress achieved in human factors can be summarized as follow:

- Better knowledge in applying the methodologies used to evaluate human factor contribution in the events and near misses events.

- Human reliability improvements in CNA I, *CNA II* and CNE reviewing normal operating procedures and emergency procedures taken into account the PSAs results.
- The Human Performance Programs of CNA I, CNA II and CNE were developed and implemented.
- Fostering the training program addressing past incorrect human behaviour and mistakes.
- Widespread personnel re-training using international and national events.

3.12.5. COMPLIANCE WITH THE OBLIGATIONS IMPOSED BY THE CONVENTION

Methods to prevent, detect and correct the occurrence of events related with human factors are being used by the Licensee of operating NPPs. In this sense, different methodologies are used by both the Licensee and the ARN to detect both human factors and organizational deficiencies as “root causes” and “contributing causes” of analysed events, as part of OPEX Feedback Program.

Additionally, PSAs carried out for CNA I, CNA II and CNE show that human corrective actions were considered to be necessary in order to ensure that the capabilities and limitations of human performance were taken into account in the procedures for both normal and abnormal operation situations.

Therefore the country complies with the obligations imposed in Article 12 of the Convention on Nuclear Safety.

3.13. ARTICLE 13: QUALITY ASSURANCE

Each Contracting Party shall take the appropriate steps to ensure that quality assurance programs are established and implemented to guarantee that the specified requirements for every activity that is important to nuclear safety are satisfied throughout the life of a nuclear installation.

3.13.1. INTRODUCTION

The establishment and implementation of appropriate Quality Assurance Programs (QA Programs) in design, construction, commissioning, operation and decommissioning stages of nuclear installations is a regulatory requirement in Argentina. For NPP, Regulatory Standard AR 3.6.1. "Nuclear Power Plant Quality System" sets the requirements *to be met*. In addition, Regulatory Standard AR 3.7.1. determines when the Licensee must present the QA Program and the QA Manual.

QA Programs and Manuals, among other documents, are also mandatory for the nuclear installations.

The Regulatory Body controls the NPPs' QA Programs implemented by the Licensee by audits carried out according to the usual methodology.

3.13.2. QUALITY SYSTEM

Since the creation of NA-SA, the Board of Directors decided to implement a Corporate Quality Assurance Program in order to provide a framework for the development of particular quality programs at each facility that are documented, implemented and revised by the Plant Management and evaluated by an independent NA-SA's *Quality* Organization.

The Corporate QA Manual describes the Quality System implemented in NA-SA. Its main objective is to set the requirements that must be met in order to ensure quality achievement and its maintenance throughout the different activities developed by NA-SA, as well as to coordinate and integrate the mission, common objectives and activities of the organization.

The Corporate QA Manual comprises the *Integrated Quality and Environment Policy* for the entire organization (see Annex VI). The three NPPs (CNA I, CNA II and CNE), *Embalse NPP Long Term Operation Project (CNE-LTO)*, Nuclear Projects Unit (*DPN*) and Maintenance Support Unit (SPC), develop their own programs according to the requirements set in the Corporate Manual.

NA-SA's Corporate *Quality System* is documented in the Corporate QA Manual, program documents, procedures and QA Manuals and procedures of internal organizational units.

QA Management is responsible for the evaluation of the implementation of the *Quality System* in CNA I, CNA II and CNE, as well as in *CNE-LTO*, Nuclear Projects Unit and Maintenance Support Unit. The results of these evaluations are reported to the highest level of NA-SA.

Periodically, the QA Management issues reports showing the audits' results. These reports are then sent to NA-SA's President of the Board of Directors, the General Manager, and the Plant Managers. The results are also informed to ARN, according to the requirements of the Operating Licenses.

Figure 3.13.1 shows the QA Organizational Chart.

The status of the QA main documents is shown in the following table:

ORGANIZATION UNIT	DOCUMENT	NUMBER OF PROCEDURES
NA-SA	Corporate Quality Assurance Manual Approved December 2015	*41*
CNA I and CNA II	Quality Assurance Manual for the Operation of CNA I and CNA II, Approved April 2014	*218*
CNE	Quality Assurance Manual for the Operation of CNE Approved *March 2019*	*206*

DPN	Management System Manual Approved *May 2016*	*75*
SPC	Management System Manual Approved March 2016	27
CNE-LTO	Management System Manual Approved May 2015	*192*

QA Programs for NA-SA, CNA I and CNA II NPPs, CNE NPP, *CNE-LTO*, DPN, *meet* the requirements of Regulatory Standard AR 3.6.1. and IAEA Practice Code 50-C-Q and its applicable Safety Guides.

The DPN Management System Manual complies with Regulatory Standard AR 3.6.1, *IAEA Application of Management System for Facilities and Activities GS-R-3*, ISO Standard 14.001 Environmental Management System, ISO Standard 9001 and Standard *OHSAS 18001* Occupational Health *and Safety Management System Specification*.

Annex VI shows NA-SA's Integrated Quality and Environment Policy.

The Licensee verifies that every person and organization involved in the nuclear area becomes thoroughly familiar with QA requirements. By means of qualification and training, the objective of making the personnel familiar with new concepts and safety requirements is achieved.

In addition to the dissemination of the general principles of quality and safety culture, the involved staff is trained every time a new procedure is approved or a new revision performed. This practice, usual in the installations, is being intensified in the support sections of the Licensee.

Temporary personnel receive specific qualification and training before starting their duties. In addition, their performance is mainly evaluated when they carrying out tasks related to programmed maintenance tasks. The Licensee's staff supervises tasks carried out by temporary personnel.

In October 2018, NA-SA achieved ISO 9001:2015 certification for its Corporate QA Program for the *scope:* "Corporate management for electrical energy generation and nuclear projects", valid until *September 2021*.

In the same way, in August 2018 NA-SA *certified its environmental management system according to ISO 14.001: 2015*. Its scope comprises electric power generation at CNA I; electric power generation at CNA II; electric power generation and Cobalt 60 production in CNE; activities related to *CNE-LTO Project; management of nuclear projects by DPN Unit*; technical and administrative management at Headquarters. The Environmental Management System was first certified in August 2003.

At the end of 2015, NA-SA began the gradual, progressive, and sustainable implementation of a corporate Risk Management System. During 2016, processes were evaluated, developing matrices of corporate risks and process maps. In 2017, the second round of process evaluation and follow-up of the action plans committed during 2016 continued. Simultaneously with the certification of the 2015 versions of the ISO 9.001 and 14.001 standards in 2018, the risk management was integrated to quality and environmental management systems. Also in 2018, WANO recommendations related to risk management in the NPP operation were incorporated.

ARN audits the QA Programs of the above mentioned NPPs following the corresponding procedure. The Quality System programs must meet the Regulatory Standard AR 3.6.1., "Nuclear Power Plant Quality System", (consistent with IAEA Code 50-C-Q), the Operating License requirements and any other requirement on this subject issued by the ARN.

3.13.2.1. QUALITY SUPERVISION OF EXTERNAL SUPPLIERS RELATED TO OPERATION, REFURBISHMENT AND CONSTRUCTION OF NPPs

All new SSCs acquired for the plants' construction, refurbishment, design modifications or replacements due to obsolescence must meet the specified requirements which, in some cases, may be stricter than those in the original design.

A first step to ensure the quality of new SSCs is the adequate preparation of the technical specifications. For replacement of safety critical SSCs, technical specifications are acquired directly from the original designer.

The next step is the proper selection of suppliers. Specific procedures are applied to evaluate the ability of a supplier to supply a particular good or a service. These procedures foresee inspections at the supplier's facility and review of their quality system before awarding the contract.

CNA I and CNA II

Changes to the installation and documentation, both permanent and temporary, are reviewed by individuals or working groups other than those who developed the original work. After this, the corresponding projects must be approved by the Site Manager. According to a defined hierarchy from the safety point of view, the Technical Review Committee (CRT) and the Regulatory Body intervene for final acceptance. The methodology for this process is defined in a procedure.

For the procurement of SSCs and the services involving changes in the plant design, the methodology established in "Technical Specifications" document is used. The NA-SA procedure "Assessment of the ability of a supplier to supply goods or provide a specific service" is applied.

CNE-LTO Project

Within the framework of the CNE PLEX, NA-SA has made a major effort to develop domestic suppliers of critical SSCs for CNE´s refurbishment. In order to qualify domestic suppliers, international standards have been applied and it has been hired the original designer's supervision. The qualification process includes in some cases the preparation of a pre-production small number and subsequent testing to demonstrate that the SSCs comply with the established requirements. Only after the SSC´s preproduction series is accepted and the qualification is achieved, is authorized the production. This process has allowed the qualification of domestic companies. During the SSC´s manufacture, when the manufacturing is done in the country, inspections and audits are performed to the supplier. When deemed appropriate, a permanent resident inspector is set at the factory.

Quality programs audits are performed to suppliers in order to confirm that the supplier quality program is implemented and meets the established requirements. The main processes during manufacturing are also audited. When components are manufactured abroad, widely experienced suppliers are selected, recognized as suppliers of such components for the nuclear industry having international qualifications. In certain cases a third party can be delegated to audit or monitor on behalf of NA-SA during manufacture stage.

3.13.3. QUALITY MANAGEMENT PROGRAM OF THE CAREM PROJECT

The QA Manual describes the Quality System implemented in CAREM Project. Its main objective is to set the requirements that must be fulfilled in order to ensure quality achievement and maintenance throughout the different activities developed by the Project, as well as coordinate and integrate the common objectives, the mission and activities of the organization.

The Quality Manual comprises the Quality Policy for the entire organization. CAREM Project's QM General Program is documented in the Quality Manual, together with the procedures, general documents, Quality Plans and internal procedures of the organizational areas.

Quality Management Department is responsible for evaluating the implementation of the QM Program in CAREM Project's areas. The results of these evaluations are reported to the highest level of Upper Management.

Periodically, the Quality Management Department issues reports showing the audits' results.

These reports are then sent to Upper Management and the Areas Manager.

The Quality Manual was approved in 2009 and there are 35 implemented quality procedures.

The Quality Manual is aligned with the requirements of AR 3.6.1., *and ISO 90001:2015,* takes into consideration IAEA *GSR Part 2,* IAEA GS-G 3.1 and NCA 4100 from section 3 of the ASME B&PVC. *The Quality Manual is issued in its fourth revision. Additionally, CAREM Project complies with CNEA's Quality and Environmental Policies.*

Among the main quality procedures developed, it is worth mentioning:

- Design control.
- Design changes control.
- Construction changes control.
- Supplier's qualification.

- Supplier's evaluation.
- Special processes control.
- Product quality level determination.
- Classification of SSCs.

The design authority of the project relies on the President of CNEA, who in turn delegates the execution of the actions for the concretion of the project to the project's upper management.

Among its responsibilities, the design authority shall assure the configuration control.

The configuration control process is implemented for assuring the integrity of the plant throughout its life cycle. In the design stage, procedures were put forward to plan, execute, verify and validate the design taking into account the fulfilment of the requirements for the SSCs. Design changes are also controlled in the construction phase so that the purpose of the design is not altered despite changes and the compliance to the requirements is maintained. This includes temporary modifications due to, for example, assembly activities. Procedures are to be developed and implemented through commissioning, operations and decommissioning.

The organization verifies that every person involved in the nuclear area becomes thoroughly familiar with QM requirements. By means of qualification and training, the objective of making the personnel familiar with new concepts and safety requirements is achieved.

Every staff member, regardless their area of expertise, is trained on the general principles of quality, the quality program and safety culture through an induction course, along with nuclear power, regulation and project's general description lectures.

As a new practice, involved staff is trained every time a new procedure is approved or a new revision performed.

The project's quality program is annually audited following AR 3.6.1. by the ARN, including over the last three years the evaluation of the processes of design, training, and the implementation of the quality system.

Figure 3.13.2. shows the organization for CAREM Project.

3.13.3.1. QUALITY SUPERVISION OF EXTERNAL SUPPLIERS RELATED TO DESIGN AND CONSTRUCTION OF CAREM PROJECT

Through the procurement process it ensures that the inputs, outputs and services purchased meet specified purchase requirements. For this purpose the technical specifications provided to suppliers must be clear, concise, and unambiguous, with a full description of the product and should include technical and quality acceptance criteria corresponding requirements.

In order to ensure the proper selection of suppliers, specific procedures are applied to evaluate the ability of a supplier to supply a particular good or a service. These procedures foresee inspections at the supplier's facility and review of their quality system before awarding the contract.

The quality system classifies inputs, products and services in order to determine the quality requirements. This method takes into account the importance of such safety requirements.

Before providing a product for acceptance, suppliers must demonstrate that the requirements defined in the purchase order have been satisfied. The inspection and testing plans, including points of presence and stopping points, must be approved by a responsible of CAREM Project.

Utilization of quality plans for the development of a product or a service provision is one of the tools to ensure that the standards, specifications, procedures and acceptance requirements necessary to obtain the required quality are met by suppliers. In every contract, agreement, purchase order, etc., suppliers must submit relevant quality plans, previously approved by the CAREM Project, where the documents required for the product are specified.

3.13.4. COMPLIANCE WITH THE OBLIGATIONS IMPOSED BY THE CONVENTION

Adequate QA Programs for those activities which are important for safety throughout the nuclear installations life have been implemented for the Licensee. Therefore, Argentina meets the obligations set in Article 13 of the Convention on Nuclear Safety.

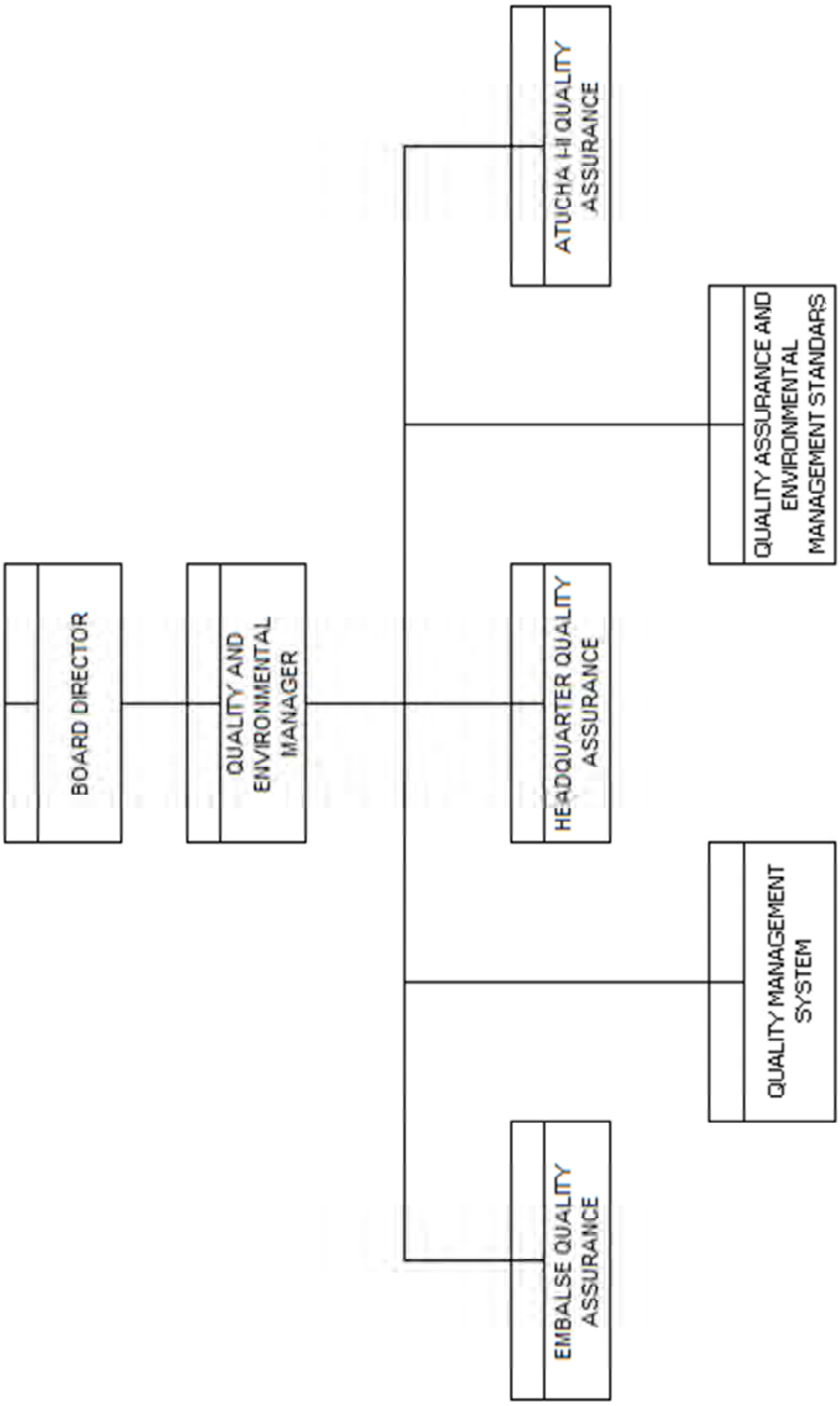


Figure 3.13.1 - QA Organizational Chart.

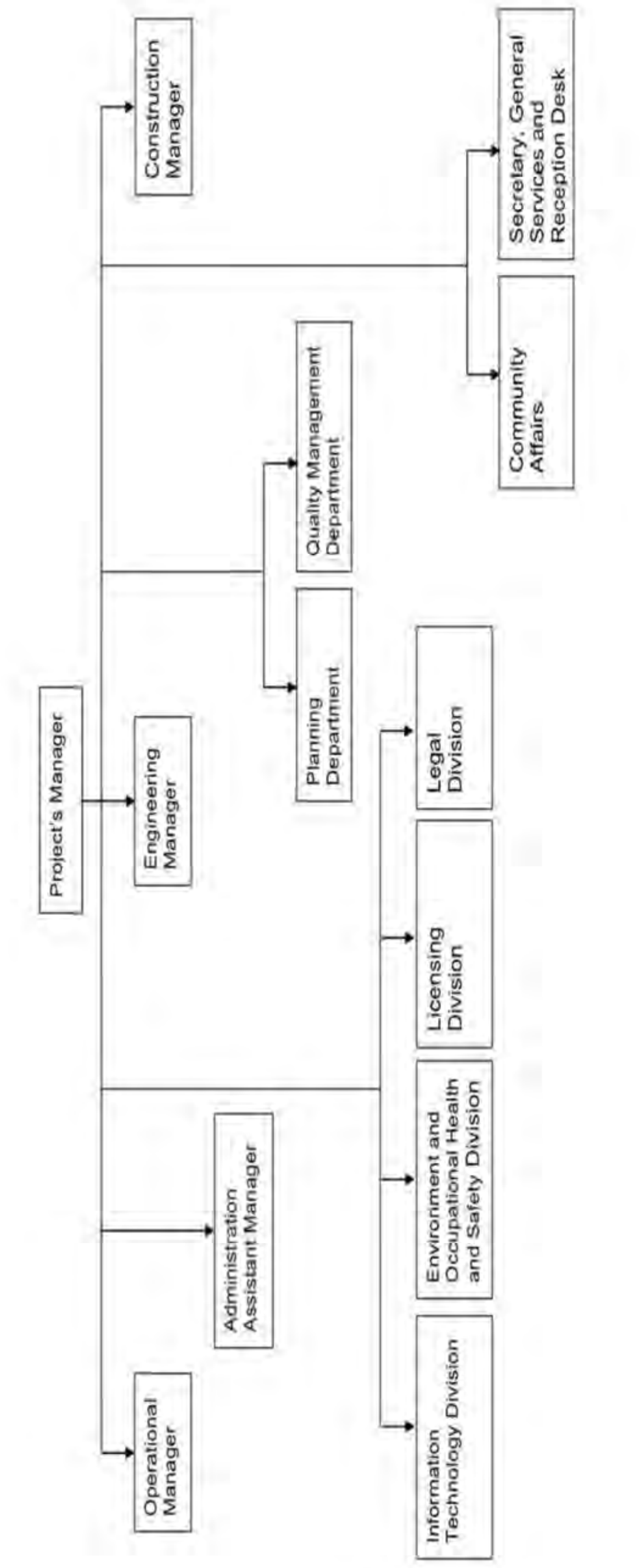


Figure 3.13.2 - Organizational Chart CAREM project.

3.14. ARTICLE 14: ASSESSMENT AND VERIFICATION OF SAFETY

Each Contracting Party shall take the appropriate steps to ensure that:

i. Comprehensive and systematic safety assessments are carried out before the construction and commissioning of a nuclear installation and throughout its life. Such assessments shall be well documented, subsequently updated in the light of operating experience and significant new safety information, and reviewed under the authority of the Regulatory Body;

ii. Verification by analysis, surveillance, testing and inspection is carried out to ensure that the physical state and the operation of a nuclear installation continue to be in accordance with its design, applicable national safety requirements, and operational limits and conditions.

3.14.1. INTRODUCTION

The maintenance of an adequate safety level in nuclear installations is a requirement of the regulatory system. Therefore, since the initial stage of a NPP project until its decommissioning, the Licensee performs different studies, either as response to an ARN requirement or as a demand of the Licensee itself. ARN controls the safety level by means of inspections, audits and assessments of the studies carried out by the installation or verifying the results by performing its own analysis.

Safety assessment and verification constitute the basis and technical support of the regulatory control. The need for carrying them out comes either from the inspection and audit results, or assessments performed by the ARN, or from internal and external operating experience including that coming from foreign installations.

The above mentioned safety assessments includes the periodic revision of the probable failure modes of SSCs, and their consequences as well as deterministic studies.

The SAR is an important document containing the development and results of the radiological and nuclear safety studies, and constitutes one of the most important documents required during the licensing process.

On the other hand, the licensing process begins several years before the NPP commissioning. First of all, pre-operational studies are performed aiming at evaluating the interactions between the installation and the environment. These studies include evaluations of the site's meteorological, geological and hydrological characteristics as well as the human activities in the zone of influence of the installation (see article 3.17.). Its results mainly contribute to identify the initiating events, either natural or man-induced, to evaluate the radiological consequences of those accidents postulated in the safety analysis, to elaborate an emergency plan and to determine discharge limits of liquid and gaseous radioactive effluents of the installation. This information is then compiled and documented in the preliminary and final SAR.

Furthermore, safety assessments are usually accomplished in order to analyse the *cumulative* impact on safety caused by relevant modifications to the installations, by the occurrence of significant events within the design bases or in an extended conditions or by any other justified reason.

For each of the existing NPP, the safety assessments made during the design, construction, commissioning and operation stages up to *2016* were summarised in previous Nuclear Safety Convention reports.

The main evaluations done during the period corresponding to this Convention are shown in the following Sections.

3.14.2. AGEING

The *Integrated* Ageing Management Program (*IAMP*) is useful in preventing and detecting systematically any degradation that involves equipment, systems and components by affecting the design safety margins.

Each NPP in operation has implemented its own *IAMP*, required by ARN with the objective of maintaining the safety of the NPPs' lifetime by optimising the inspection and maintenance programs, as well as, the monitoring, prevention and mitigation of ageing.

The monitoring of components degradation is included within the *IAMP*. It is based on evaluation of the operating conditions for each component under analysis, such as working temperature, irradiation field, component functional characteristics and human errors that can lead to worst normal degradation conditions.

In order to have effective IAMP, the following aspects based on IAEA SSG-48 were considered:

- *Scope setting in order to identify SSCs subject to ageing management*.
- To study *for* the selected components *the* ageing mechanisms, including *service conditions, degradation sites, degradation mechanisms and ageing effects*.
- *Detection of ageing effects including the technology for detecting ageing effects before the failure of structures or components.*
- *Monitoring and trending of ageing effects.*
- *Mitigation of ageing effects through operations, maintenance, repair or replacement.*
- *Clear definition of acceptance criteria against which the need for corrective actions is evaluated.*
- *Consideration of operating experience and feedback from researches. Also, consideration of IGALL experience as compiled in IAEA SRS 82.*
- *Corrective actions when the acceptance criteria are not met.*

As it was previously stated, well known methodologies are used to evaluate such degradations:

- Predictive and preventive maintenance.
- In-service Inspection program.
- Additional testing.
- Degradation evaluation.
- Evaluation of acceptance criteria in each case.

The Licensee have included the *IAMP* within a Life Management (PLIM) Program to analyse the life extension for both CNA I and CNE plants, considering that the safety of a NPP is a necessary but not sufficient condition for life extension (PLEX) *or justification of continued operation*. These programs consist in an integration of ageing management and economic planning to maintain the safety level and maximize return of investment over the service life of the plants.

In the case of CNE during the PLEX Phase I, condition and life assessment were performed for all SSCs important to safety. The outcome of these assessments was the identification of needs of replacement, refurbishment or the prognosis for the SSCs to maintain functionality keeping safety design margins during the new cycle of operation. According to these results, a feedback to the ageing management program was done and a set of recommendations emerged to modify the preventive maintenance, *test procedures program* and the ISI *program* to be implemented during the second operation cycle.

Both the Licensee and the Regulatory Body have been *(and still continue)* carrying out several training activities (participation in IAEA Pre-SALTO and SALTO workshops, IAEA IGALL meetings, workshops with AREVA , EPRI and NRC, etc.) in order to improve the technical capacity on ageing assessment.

The main activities performed in the reported period related with ageing management of SSCs are the following:

CNE:

- *The improvement of the Ageing Management Program (called MGE-CNE and based on IAEA Specific Safety Guide SSG-48 "Ageing Management and Development of a Programme for Long Term Operation of Nuclear Power Plants", 2018) for the second cycle of operation using the lessons learned from the condition and life assessments performed at individual SSCs.*
- *Also, some plant programs like maintenance and ISI were improved for the second period of operation.*

- *Regarding management of obsolescence, an Ageing Management Program was developed based on IGALL AMPs.*
- *In addition, a methodology for the surveillance of the containment structural integrity was developed based on the leak tightness test done as a pre-condition for the reactor fuel loading. A base line result was obtained from the leak test which will be used for monitoring the tendon's loss of stress during the second period of operation.*

CNA I and CNA II:

- *Systematic scoping and screening was completed following safety classification of SSCs and the items a), b) and c) of paragraph 5.16. of IAEA SSG-48.*
- *In the case of CNA I*, a procedure for the ageing management review as well as for condition assessment was agreed between the utility and ARN.
- *Besides, a* comprehensive degradation mechanism matrix for specific materials and water chemistry of CNA I / II was developed.
- A list of Time Limiting Ageing Analysis (TLAAs) *for CNA I* was submitted by the utility to ARN. Special considerations were taken for the RPV and its internals.
- In the case of CNA I, it was also developed the condition assessment for almost 80% of the SSCs.
- IGALL recommendations were taken into account.
- The management of recommendations raised from condition assessment reports is in progress. The recommendations were assigned to the primary responsible areas.
- *The plant programs like maintenance are being reviewed against the nine attributes of IAEA SSG-48.*
- *Regarding management of obsolescence, an Ageing Management Program was developed based on IGALL AMPs.*

3.14.2.1. CNA I REACTOR PRESSURE VESSEL

CNA I started commercial operation in 1974. The base material of the RPV is similar to those of other NPPs RPV at that time, low alloy ferric carbon steel equivalent to DIN 22NiMoCr37 and similar to ASTM A 508 class 2 forging.

The companies that participated in manufacturing the RPV were the following:

- Manufacturing of the steel base material, forging and vertical welding: Rheinstahl Heinrichshüte.
- RDM *(Dutch manufacturer) performed some assembly welds*.

The acceptance *examinations and* tests performed in the manufacturing process were the following:

- Ultra sonic testing, SIEMENS procedure.
- Tensile tests *and* Charpy V notch impact tests.

Initially, KWU (the Designer), didn't consider it necessary to formulate a surveillance program for the RPV's material because a very low fast neutrons fluence (E>1 MeV) was estimated in the beltline region, for it to produce important changes in the material brittle to ductile transition temperature throughout the CNA I's design life time (EOL- end of life, corresponding to 32 years of full power operation). Nevertheless, the Licensee required the designer to formulate a surveillance program for the RPV which was undertaken by KWU in April 1974 during the CNA I commissioning.

Due to design constraints of the RPV, it was not possible to implement a surveillance program *representative of the beltline*, but several irradiations programs were performed in order to evaluate the embrittlement of the belt line material.

Although those results showed that the RPV should safely operate until EOL, the Regulatory Body required to the Licensee additional safety studies.

These studies consisted in performing further irradiations and testing more samples, and making new evaluations on neutronics and Pressure Thermal Shock (PTS) subjects. For these tasks the Licensee received the advice of KWU/Siemens, SCK- Mol and VTT, international experts.

The above mentioned studies also included material irradiation up to 1.5 EOL fluence in order to be able to justify a possible Life Extension.

The results of the activities concluded that the integrity of the CNA I's RPV is guaranteed until EOL and for 1.5 EOL, material properties could be sufficient to *support* the consequences of the most demanding transient. As AR standards do not have specific acceptance criteria for PTS, those defined by the French and German guides were applied to the results of the irradiation programs.

ARN considers that, taking into account the results of the evaluations performed, the RPV structural integrity is assured up to EOL and all the in service inspection *(ISI)* results show that there is not *relevant* indications that could jeopardize the integrity. Nevertheless, the RPV integrity assessment is considered a permanent issue and for that reason, ARN considers necessary a permanent evaluation in the areas of non-destructive examinations, PTS and neutronics, including the results from the periodic RPV inspections as well as the possible improvements in the evaluation techniques as a way to reduce the uncertainties and to improve the general knowledge on this subject.

RPV´s inspections are included in the ISI Program and are performed according to ASME *Code, Section* XI code in terms of the areas, frequency and with the scope practicable for this PHWR's RPV. The last ISI campaign was in 2018, no relevant indications were reported so far. *Since 2015 different elements of the RPV (following the ISI program) were inspected during the outages 2016, 2017 and 2018, and no relevant indications were found.*

In spite of the definition of the inspection areas follows the ASME code rules, considering the experience gained with the events occurred at Doel 3 NPP and Thiange 2 NPP (flaw indications detected), the ARN and the licensee agreed to enlarge the examination volume corresponding to ASME Code to that area where quasi-laminar indications were found in the above mentioned NPPs. This experience yields the decision to inspect the base material in the upper and lower core shells.

During the 2013 outage, the inspection was made by automated ultrasonic and carried out jointly by Doosan Power Systems and the licensee and covered all plates in the RPV cylindrical portion, no indications were detected. The examination volume and the defect acceptance standard have been defined by the licensee following the indications found in the Belgium plants:

- 100% examination of the full wall thickness.
- Inspection of six of vertical segments of the RPV in a full length and 700 mm wide.
- Target flaw size: 10 mm x 10 mm quasi lamination defects are oriented parallel to the inner and outer surfaces of the RPV.

The results of the inspections showed that no *relevant indications* have been detected within the inspection volumes of any of the six segments of base material.

Considering that ARN was informed in 2014 about the utility decision to face a long term operation project, and taking into account that the structural integrity of the RPV is a TLAA, ARN was proactive, in defining the activities that should be done to build confidence that the RPV is fit for future service.

To do so, ARN was supported by a team of international and national experts. The activities identified and *done* by the utility are:

- Transients analysis update to support the most severe one for the RPV wall.
- Identification of the new belt line region (as NRC definition).
- Structural integrity analysis for the inlet coolant nozzle, considering degraded material properties.
- Use of less conservative and more realistic material embrittlement correlations.
- *Update of the neutron calculations and confirmation* of the fast fluence *1.5 EOL*.

Regarding the revalidation of the resistance to brittle fracture of the pressure vessel, NA-SA delivered a technical report containing different evaluations, all according to the state of the art in what refers to the use of empirical correlations for the estimation of the irradiated material's properties.

Likewise, NA-SA contracted CIEMAT (Institute of Spain) to carry out an independent review of the report before its submission to ARN.

Additionally, NA-SA presented the fracto-mechanical study of the pressure vessel against the occurrence of a severe transient, such as pressurized thermal shock, postulating the existence of different types of cracks, orientations, locations and for the load conditions of the postulated transient.

The regulatory review performed by ARN, allowed to conclude that the integrity of the pressure vessel is assured for 1.5.EOL (1.89.1019 n / cm^2 (E> 1MeV)). So, the critical component is fit for additional operation maintaining a sufficient safety margin during the long term period of operation.

3.14.2.2. CNE TESTING OF SAFETY RELATED COMPONENTS

As it was mentioned in Section 3.6., CNE performed the refurbishment outage during which the integrated implementation plan was satisfactory implemented and several design upgrades were performed, for facing the life extension period.

During the first life of operation several testing activities were performed in order to confirm availability and functionality of safety related structures and components.

In general, these activities were performed following the applicable industry standards like Canadian Standards Association, CSA. For example, requirements for testing and acceptance criteria that were used in CNE are: CSA N285, 4 “Periodic Inspection of CANDU nuclear power plant components”; CSA N285, 5 “Periodic Inspection of CANDU nuclear power plant containment components” and CSA N287, 7 “In service examination and testing requirements for concrete structures for CANDU nuclear power plants”.

The main critical components that were in scope for these activities are pressure tubes, feeders and steam generators.

In the case of pressure tubes the activities included garter springs repositioning between pressure tubes and calandria tubes, inspection and scrapping in order to determine the hydrogen equivalent concentration.

While in the previous National Reports these activities were described in detailed, in the reported period of the present Report there were no activities in this field due to the fact that during the retubing campaign (2016 to 2018) all the pressure tubes and calandria tubes were replaced by new ones with improved design features.

Garter springs were changed for new ones with an enhanced design focused on avoiding the displacement or movement. Basically, the new design shows square wire profile and is tight to the pressure tube. Experience in previous refurbishment showed that this design has been successful not showing any movement.

Regarding inspection in pressure tubes, the activities during the reported period were only those performed after the pressure test as part of the construction process of the new pressure tubes. The results of these inspections constitute a base line for supporting the inspections campaigns that will be done during the second life of operation.

The Licensee will perform dimensional and flaw detection inspection campaigns for assurance of fitness for service during the continued operation.

Also, all feeders were replaced on the refurbishment outage. The new design incorporates a material change assuring a minimum of chromium content. This was in order to improve the behaviour regarding the flow assisted corrosion and stress corrosion cracking degradation. It is foreseen that nor wall thickness reduction neither crack at feeder bends, be an issue during the second life of operation.

Steam Generators

The original Embalse Steam Generators (SGs) are BWXT (formerly Babcock and Wilcox Canada) design; they were vertical-recirculating heat exchangers with alloy I-800 inverted U-tubes, internal preheater and cyclonic steam separators. The internals (tube support plates, U-bend supports, shroud and steam separators) were carbon steel.

Along the operation years Embalse SGs have experienced ageing mechanisms as Flow Assisted Corrosion that affected carbon steel internals (tube support plates, u-bend supports and the primary steam separators) that limited the component life and lead to a fitness-for-service program implemented during the last operating years of the design life. Twin plants as Pt. Lepreau and Gentilly 2 did not experience this issue since material selection for the tube support plates was stainless steel.

Under the Embalse Plant Life Extension program, the replacement decision of the 4 SGs was taken on 2007. The Embalse steam generator replacement involved the lower assembly or “Cartridge” replacement, and the steam drum in-situ refurbishment for reuse. The scope and methodology of the activity were defined based on facts and factors that are not adjusting variables, such as: CANDU 6 plant design did not take into account replacement of large components and a temporary opening of the reactor building was not feasible.

The design changes for the new SGs mainly respond to the following factors based on the Qinshan project reference, which is an improved CANDU 6 steam generator design:

Plant repowering:

- *Power uprating from 2015 MW(t) to 2064 MW(t) required a heat transfer surface increase from 2350 m^2 to 3195 m^2.*
- *Feedwater temperature increase from 167Cº to 187ºC.*

Seismic re-qualification (Minor Modifications category):

- *Seismic qualification increase from 0.15 g to 0.27 g.*
- *Addition of snubbers to the lower lateral supports.*

Design changes from operational experience:

- *New design and material of the TSPs and the U-Bend supports. The new TSPs´ design is "flap bar" type and the adopted material is stainless steel.*
- *New inspection port at the region of the U-bend.*
- *New inspection ports for access to all tube support plates.*
- *New inspection ports at the preheating region.*
- *An additional inspection port at the top of tube sheet for water lancing.*
- *Additional water level nozzles for the increase of the safety system trip coverage (SDS1 and SDS2).*
- *Integral primary-head divider plate.*
- *New carbon steel pressure boundary components in contact with fluid (wet surfaces during service), have a minimum of 0.2% Cr on their chemical composition.*
- *New stainless steel primary cyclonic steam separators.*

The main milestones of the project were:

- *Plant shut down for refurbishment in December 2015.*
- *New steam generator Cartridges delivered to site in August 2016.*
- *Steam drum cutting completed in November 2016.*
- *Steam separator replacement completed in December 2016.*
- *New four Cartridges fit-up completion in June 2017.*
- *Steam drum re-welding completed in January 2018.*
- *Primary Heat Transport System pressure test completed in October 2018.*
- *Secondary Side System pressure test completed in November 2018.*

3.14.3. SAFETY ASSESSMENT

Safety assessments cover all the plant operating modes, evaluation of design bases and design extension conditions events and include a periodic revision of failure modes of SSCs, identifying the consequences of such failures as well. For old plants some original operation safety criteria were different from those used nowadays, thus it is necessary to make an additional effort in order to take into account the application of new safety criteria.

In Argentina two complementary methods are mainly applied in safety assessment: the deterministic and the probabilistic one.

The main deterministic and probabilistic assessments done during the period corresponding to this report are shown in subsections 3.14.3.1. and 3.14.3.2.

3.14.3.1. DETERMINISTIC ASSESSMENTS

Deterministic Assessments are performed mainly focused on the Safety Analysis for each NPP and it is included in the Safety Analysis Report (SAR) which is a mandatory document elaborated by the licensee in order to obtain a license or for a periodic safety review. In addition to this, some deterministic assessments are performed as support for the Probabilistic Safety Analysis, for Severe Accident Analysis and for specific requirements, such as the stress test analysis after the Fukushima accident.

3.14.3.1.1. Safety Analysis Updating

In the period 2013-2016, a review of the SAR of CNA II considering the experience gained during commissioning stage was submitted to the Regulatory Body, in order to obtain the Operating License. In May 2016, the Operation License was granted to the CNA II.

According to Argentinean Regulatory Standard AR 3.9.1. –"General Criteria for Operational Safety in NPP" the SAR of a nuclear installation must be updated each time that a plant design modification is performed, and once every five years. So, the updating of CNA II's SAR is scheduled for 2021.

An updating of the CNA I´s SAR was performed and submitted to Regulatory Body for License Renewal in September 2014. In September 2017 a new version of SAR Chapter 15 "Accident Analysis" was released. This document presents the results of simulations performed with updated methodology and calculation tool and was one of the key documents by which the Licensee demonstrated the fulfilment of the licensing basis for facing the Phase A of the long term operation (see Section 3.6. for more detail).

Licensee *performed* the upgrade of CNE´s SAR which *was* completed as a condition for plant restart after the refurbishment outage. This upgrade includes the design changes implemented for life extension.

Regulatory Body is evaluating the deterministic safety assessment (DSA) of CAREM 25 prototype reactor.

3.14.3.1.2. Post Fukushima Analysis

As a consequence of the Fukushima accident and with the purpose of applying the corresponding lessons learned, the Regulatory Body requested to perform a stress test to each Argentinean NPP which, among other safety related issues, included:

- Loss of Offsite Power (LOOP).
- Station BlackOut (SBO).
- Loss of heat sinks.
- Loss of heat sinks coincident with SBO.

As was mentioned in Chapter 2, as result of the evaluation several improvements were required by the Regulatory Body. Those of them implemented before *2016* were detailed in the *Previous* National *Safety Reports.*

Deterministic assessments were developed or are under development in order to support the elaboration of the Severe Accident Management Guides (SAMG) according to the strategies proposed in the framework of the Severe Accidents Management Program.

The SAMG that were already issued are detailed in Section 3.14.3.1.3.

Design changes implemented in CNA I and CNA II during the period *2016-2019* are explained in Section 3.6.

Design changes implemented during refurbishment outage of CNE NPP are showed in Section 3.6.5.

3.14.3.1.3. Accident Management and Severe Accident Management Program

As was mentioned in previous CNS Reports, the ARN required the Licensee to develop a Severe Accident Management Program (SAMP) for the plants in operation. The initial activities were:

- Development of internal procedures for the program (Organization, QA, etc.).
- Selection of a methodology to characterize plant damage scenarios.
- Selection of six plant damage state scenarios to be characterized during the first stage of the program.
- Starting the development of a model for severe accident progression. MELCOR package was selected.

The program execution continued with more specific activities, as follows:

- Plant damage states characterization including deterministic studies related to new scenarios (or extending the existing ones) with RELAP code up to core uncovered initiation.

- Strategies proposal (preventive strategies for the time).
  - Small LOCA plus failure of low pressure emergency water injection.
  - Small LOCA plus moderator pumps failure which delays low pressure emergency water injection due to depressurization delay.
  - Blackout (SBO).
  - Unavailability of the second heat sink pumps or lack of sufficient inventory for different accidental scenarios.
  - Analysis of potential strategies related to both the use of an alternative control room in case the main control room becomes uninhabitable and the possibility of venting the containment.
- Progress in the development of the accident progression model. The experience of CNA II PSA Level 2 was transferred back to CNA I SAMP.
- In the same way, the experience gained for CNA II with Passive Autocatalytic Recombiners (PARs) was used in CNA I applications.

A schedule to continue CNA I SAMP development was approved by ARN in 2009. The activities related to CNA I SAMP are those related to plant damage states characterization, proposal of strategies and definition of the methodology for the development of the instructions and guidelines. Those tasks are being closely followed by ARN personnel.

As part as the stress test, it was required to each Argentinean NPP to perform a reassessment of the NPPs safety margins assuming the occurrence of a sequential loss of the lines of defence in depth caused by severe accidents and, among others safety related issues, includes:

a. 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; after fuel damage has started and, after RPV has failed).
b. Describe accident management actions and plant design characteristics to protect the confinement function integrity after fuel damage has begun (prevent hydrogen explosion / deflagration; overpressure prevention, re-criticality prevention; containment flooding to prevent RPV failure or to limit the molten core-concrete interaction –MCCI-; and need of alternative electric power).
c. Describe accident management measures presently available to face the successive steps in a scenario of failure of spent fuel pool cooling (radiation protection, top of fuel uncover and, fuel degradation.
d. Identify any possible cliff effect.
e. Evaluate the adequacy of accident management strategies (guides and procedures developed to face a severe accident; analyse the possibilities of additional actions -adequacy/availability the required instrumentation; habitability and accessibility of essential areas and; hydrogen accumulations in buildings different from the containment).
f. Organization (staffing, resources and management shifts; use of external technical support and; procedures, capacitation/training and exercises).
g. Availability to use existing equipment.
h. Forecasts for the use of mobile equipment.
i. Availability and supply management.
j. Management of radioactive emissions and forecasts possible to limit them.
k. Management of potential doses to workers and provisions to limit them.
l. Systems of communication and information.
m. Activities planned for the long term (after the accident).

In response to the mentioned regulatory requirement the CNA I, CNA II and CNE Licensee performed the required stress test and submitted to the Regulatory Body the corresponding Stress Test Reports. Later on, the Regulatory Body carried out an assessment of these reports to verify compliance with the provisions of the regulatory requirement. As a result, many opportunities for improvements related with severe accident management and recovery have been identified but, in no case, weaknesses which make necessary to take urgent actions were showed.

The improvements and modifications proposed by the NPPs Licensee included an implementation schedule composed by short, medium and long term actions. The improvements as the stress tests results include the following:

3.14.3.1.3.1. CNA I

Severe accident management program

The Severe Accident Management Manual (applicable to Units I and II), which gives structure and mechanisms to develop the Severe Accident Management Program and main guidelines for the management of a situation of accident, has been developed.

Procedures have been developed for Units I and II, regarding the organization of the program, generation / control / file / distribution of documentation and their technical adequacy.

Several guidelines have been developed within this program. For more detail about development of instructions and guidelines see section 3.14.3.1.3.5

Plant Model for Severe Accident simulations

As was mentioned in previous CNS Reports a *preliminary* accident progression model for CNA I was developed using the MELCOR code, including containment failure or by-pass after a core meltdown and the evaluation of proposed mitigation strategies effectiveness.

Later it was decided to perform a "best estimate" analysis, and for this purpose it is more appropriate to use the RELAP5 / SCDAP code. *RELAP5 / SCDAP* was designed based on RELAP5 / mod 3.2 *code (which is* more advanced than RELAP4 version). *In addition, the code has* the capability to model core meltdown, relocation of components, oxidation, re-flooding of the degraded core and other phenomena associated with severe accidents.

The program RELAP5 / SCDAP represents in more detail than the code MELCOR both the thermohydraulic behaviour of the Plant and the in-vessel phenomena (inside the reactor pressure vessel).

Given the particular characteristics of the reactors of Atucha type, NA-SA together with *ISS*, the *current* developer of RELAP5 / SCDAP, *have developed* a version of the code that can represent the expected phenomenology in Atucha reactors (RELAP5 / SCDAP Mod 3.6). *Improvement and assessment activities of the code is an on-going task that is permanently performed by NA-SA and ISS personnel.*

In addition to reproducing in more detail the complex behaviour expected during a severe accident, RELAP5 / SCDAP is used to evaluate severe accident mitigation strategies.

Using this new code base SBO and small LOCA simulations of accident scenarios without safety injection (without countermeasures) were performed.

Progress in the development of the accident progression model

Analysis of Severe Accidents is performed with RELAP5 / SCDAP Mod 3.6 Code. Improvements in the input models, code assessment and development of new code features together with Code providers are an on-going task.

Main tasks performed during previous years consist of verification of Secondary side Feed and Beed and Primary Side Feed and Bleed severe accident management strategies.

Full PSA Level 2 and 3 has been requested by the ARN for Atucha I NPP. Therefore, oncoming tasks include the development of a full plant plus Containment MELCOR Model, calculation of Source Terms and Dispersion analysis with MACCs Code. This is a long term task that is planned for several years of work.

In parallel, the multi-compartment model of Atucha I Containment for the GOTHIC 8.1 code has been improved and updated. The model has been used for DBA to assess pressure and temperature ambient conditions for the Life Extension Project.

This model is currently being used to asses some preliminary SAM Strategies for Containment. In the future, it can also be useful to establish the technical specification of a Filtered Containment Venting System (FCVS) for Atucha I.

An Atucha I SFP 3D model for the GOTHIC 8.1 code has been developed and simulations of the loss of the SFP (Spent Fuel Pool) cooling pumps were performed, with and without gates between pools in place, considering in the latter case the water mixing between the pools. These calculations are updated annually.

Passive Auto-catalytic Recombiners (PARs)

The assessments to determine the specifications, location and quantity, detailed engineering and required procedures of the PARs were performed. These tasks were performed by AREVA. The planned installation of 32 recombiners was completed.

In the future, MELCOR Full plant model of CNA I will be used to assess PAR performance during the full set of Severe Accident Scenarios identified for PSA L2. This analysis can be complemented by detailed GOTHIC 8.1 calculations if the need of further information on 3D phenomena arises after lumped-parameter analysis.

Strategies to reduce the containment pressure during severe accidents

Guidelines *1-GAS-CE-10, 1-GAS-SC-10-1 and 1-GAS-SC-10-2* “Control of containment conditions” was drawn up, containing several strategies to reduce the containment pressure during a severe accident.

Filtered Containment Venting system

In order to evaluate the need of a filtered containment system, a two steps assessment should be performed. Firstly, it should be demonstrated through a suitable assessment that, for any of the extreme events considered in the frame of PSA L2, the maximum pressure that can withstand the containment would not be exceeded and that it can continue to fulfil its safety functions under severe accident conditions. The second step should be to install a filtered containment venting system if, from the assessment mentioned before arises that, as result of the occurrence of any of the extreme events considered, its maximum pressure would be exceeded.

Comparative analysis with PSA L2 of CNA II NPP results, show that the retention of molten material in vessel, or the stabilization of molten material inside containment is a major concern that has to be assessed firstly. If neither of these two possibilities is successful, containment pressurization is not expected, and therefore, the decrease of expected dose to the public due to the installation of a Filtered Containment System is difficult to justify.

The filtered venting system was foreseen for CNA I will be implemented in the framework of the tasks for plant life extension. Presently, all the efforts are concentrated in verifying the stabilization of molten material and the project advancement is tied to results of such analysis, which are not yet conclusive, given the difficulty of the task.

Cooling of the RPV external side

The RPV external side cooling is considered as a means for retaining the corium in scenarios with extensive core damage. The strategy and its effectiveness are currently under analysis. In the past years, preliminary results obtained with RELAP5 / SCDAP were performed. These simple calculations were followed by more complex analysis with ANSYS / CFD code, performed for CNA II NPP. The results of these analyses are not yet conclusive and therefore External Reactor Vessel Cooling can neither be totally ruled out nor considered a successful countermeasure.

Given the complexity of a conclusive answer, a parallel course of analysis has been started, to assess possible stabilization of molten material inside sump, to avoid containment breach due to molten core concrete interaction (MCCI) or at least delay it, so as to decrease consequences in public as far as reasonable achievable. This task is being performed jointly by Safety Analysis and Life Extension Project groups.

As the concept engineering of this strategy involves highly complex studies, it was proposed to incorporate it into the life extension programme.

Review and improvement of accident management procedures

The review of the following procedures has been made to ensure the operation of the systems that are necessary in the proposed scenarios for ensuring the proper functioning and demand of safety systems which are required in extreme events for at least the initial 72 hours:

- SBO: Manual action to inject the SHS in a short period of time with a cooling ramp of 100℃/h and manually deactivate the TB.
- Preventing that air from the boron injection system (TB) discharge into the PHTS in case of a SBO scenario, with potential degradation of the system cooling (natural circulation interruption) and effects on the fuel element oxidation rate. An instruction for the Blackout condition was drawn up; it is contained in the Operations Manual, in which isolating the branches of the boron

injection system (TB) is considered (the pressurization valves between the gas tank and the boric solution remain closed) before the depletion of the 24 Vdc batteries. In the particular case where the introduction of 70% of the control rods does not occur before 3 s, an automatic Boron Injection signal is triggered and the Intercept Valves of the three branches have to be closed due to the actuation of a signal, as these are fed from the bars without interrupting the supply.

- Inventory reposition of the SHS with increase in the capability of the SHS feed water tanks, using the two pumps of the conditioning water system and replace water in those pools with groundwater using one of the pumps of the potable water supply.
- Low river level. This will allow systematic manoeuvres to carry a plant outage.

Instrumentation and Control

The installation of an extra level measurement in the reactor *has already been implemented.*

Besides, a level measurement was installed in the spent fuel pool, needed to implement the actions for inventory replenishment to the pools in case of total loss of heat sinks. The necessary measurements for the implementation of other strategies, such as cooling the outer side of the pressure vessel and venting the containment will be evaluated in due course.

The implementation of the means to ensure the power feed to the instrumentation corresponding to representative signals of the variables needed to monitor the status and evolution of the plant in a SBO scenario was already performed.

A complete assessment of available instrumentation and its expected behavior during Severe Accidents conditions was also performed, and a list of minimum instrumentation needed for Severe Accident Management was identified.

The installation of the necessary I&C in the reactor core and the containment to provide the information for severe accidents management, considering the environmental conditions caused by these accidents, is *expected to take place during the Life Extension Project.*

Alternative cooling mode of the DGs

The old DGs were cooled by river water cooling ensured system (UK). DGs of the new EPS are air cooled.

Disconnection of electrical loads

The new EPS was designed with a new concept of uninterruptible system. This concept consists in loads connected to short time interruptible bus bars and other loads connected to uninterruptible bus bars. The modification consists in feeding only necessary loads to these uninterruptible buses in order to increase batteries life. This improvement has been already implemented.

Refuelling machine: fuel elements integrity assessment

The SBO impact on the fuels that could be housed inside the refuelling machine was analysed with a fuel element for 24 hours after the start of the event, considering the introduction of countermeasures planned for the 8 hours after the start. The results confirm the effectiveness of the countermeasure during the period analysed. A guideline was drawn up within the framework of Severe Accident Management.

3.14.3.1.3.2. CNA II

Severe accident management program

The Severe Accident Management Manual (applicable to Units I and II), which gives structure and mechanisms to develop the Severe Accident Management programme and main guidelines for the management of a situation of accident has been drawn up.

Procedures have been developed for Units I and II regarding the organization of the program, generation / control / file / distribution of documentation and their technical adequacy.

Several guidelines have been developed within this program. For more detail about development of instructions and guidelines see section *3.14.3.1.3.5*

Plant Model for Severe Accident simulations

As was mentioned in previous National Reports, within PSA Level 2 an accident progression model with MELCOR code was developed for CNA II.

In the same manner as for CNA I, a RELAP5 / SCDAP Mod 3.6 model was developed for Atucha II. This model has been used mainly for the assessment of in-vessel phenomena, for code to code comparison between MELCOR 1.8.6 and RELAP5 / SCDAP and for Severe Accident Management verification.

Main tasks performed during previous years consist of verification of Secondary side Feed and Beed and Primary Side Feed and Bleed severe accident management strategies. Currently, injection to Primary Side by pressure control system is being assessed.

Improvements in the input models, code assessment and development of new code features together with Code providers is an on-going task that contribute to better modelling.

Passive Auto-catalytic Recombiners (PARs)

The PARs installation has already been implemented.

Filtered Containment Venting system

The assessment to demonstrate that the maximum pressure that can withstand the containment would not be exceeded and that it can continue to fulfil its safety functions under severe accident conditions is under process. Same answer as for NPP Atucha I since some filtering common to both units is being considered.

Cooling the RPV external side

The RPV external side cooling is considered as a means for retaining the corium inside the vessel in scenarios with extensive core damage. The strategy and its effectiveness are currently under analysis. In the past years, preliminary results obtained with RELAP5 / SCDAP were performed. These simple calculations were followed by more complex analysis with ANSYS / CFD code, performed for Atucha II NPP. The results of these analyses are not yet conclusive and therefore External Reactor Vessel Cooling can neither be totally ruled out nor considered a successful countermeasure.

Nowadays, very detailed calculations with ANSYS / CFD code are being performed for SBO scenario. This analysis is intended to assess if a very Best Estimate calculation can lead to successfulness of the strategy. The calculation includes both cavity flooding phase and assessment of possible RPV failure due to thermal shock, and vessel cooling during relocation, to fully assess CHF occurrence considering RPV wall ablation mechanism and wall heatup influence, and also heat generation in the whole RPV wall.

In parallel, a set of scoping calculations with ANSYS / THERMAL code are planned, to get a complete view of the possible scenarios.

Increasing DGs autonomy

Actions for “Optimizing Fuel Management for Diesel Generators", which aims to extend the operating time of emergency generators using the fuel of the auxiliary boiler, *has already been implemented*. In addition, a modification to the facility was carried out to supply fuel to weekly tanks of the diesel generators from the supply tank of the auxiliary boiler.

As they are tasks performed by areas belonging to the Emergency Response Organization (ERO), they changed from Severe Accident Management Guidelines to internal instructions of the ERO. The instructions have been developed. See section 3.14.3.1.3.5.

Switchgear building ventilation

In order to unify the severe accident guides of units I and II, it was decided that the Severe Accident Management Guideline SC 12-1 "Habitability of the Control Room" that aims to isolate the ventilation of the UBA electrical operation building in case of High concentration of radioactivity, smoke or chemicals in the outside air, belongs to the operations manual. This improvement has been implemented.

Disconnection of electrical loads

In order to unify the severe accident guides of units I and II, it was decided that the Severe Accident Management Guideline SC 04-6 A for Severe Accident Management "Extension of Power Supply Time of Batteries", belongs to the operations manual. This improvement has been implemented.

Instrumentation and Control

The I&C of the reactor core and the containment necessary to dispose of the required information for CNA II has, by design, different types of post-accident instrumentation which monitors its status during and after an accident has occurred.

Alternative power sources

The analysis of essential consumption required to face with severe accident situations caused by a SBO was developed. The following activities are being carried out:

- Study of the connections between mobile diesel generator and electric bars.
- Definition of the power and location of the mobile diesel generator.
- Preparation of the technical specifications of the mobile diesel generator.

Detailed engineering of this improvement has already been developed.

Assessment of main and secondary control room habitability

The actions referred to "Control Room Habitability", *contained within the Operations Manual* were issued in order to ensure the long term habitability of the Control Room in case of release of radioactivity material or smoke presence in outdoor air.

A verification of the design of the main and auxiliary control rooms was carried out during 2016. In addition, a qualitative risk analysis of potential internal and external events at CNA site, which could affect the habitability of the control rooms, was also carried out.

The status of the tasks required to test the tightness of the envelope of the main and auxiliary control rooms is shown below:

- Definition of control room envelope*: finished.*
- Evaluation of operating modes of the ventilation system: *finished.*
- Analysis of interference with adjacent areas: *finished.*
- Definition of acceptance criteria: *in progress.*
- *Evaluation of improvements related to increase the main control room habitability: in progress.*

Alternative water sources

The firefighting system of the construction site was identified as an alternative water reservoir. This system is currently connected to the plant firefighting system (SGA), which is the system that will be used to face a severe accident situation caused by the loss of heat sinks. Water replenishment of the spent fuel pools and water supply to the SGs in the long term will be possible through the plant firefighting system (SGA).Some plant modifications were implemented in order to allow the connection between the SGA system and the water supply system of the SGs (GHC).

The guidelines were drawn up within the framework of severe accident management (See section 3.14.3.1.3.5.):

- *Guide 2-GAS-CE- 11 "Water Injection to the Spent Fuel Pool." which includes the replenishment of water through the demineralized water supply system (GHC) or through the firefighting water system (SGA).*
- *Guide 2-GAS-CE- 05 "Feed and Bleed of Steam Generators", which includes the replenishment of water through the demineralized water supply system (GHC).*

Reposition of light water to the primary system from the volume control system

The objective is to have a volume control system (KBA) tank loaded with light water to replenish the inventories of the primary moderator system. Water is extracted from the storage tank by means of a circulation pump that injects into the common suction manifold of the high pressure charging pumps *and, through one of them, it is sent to the main moderator system. In normal operation, the tank will be isolated. It has already been implemented and written.*

3.14.3.1.3.3. CNE

Safety system trip parameters

Improvement in the safety system trip parameter coverage *was* foreseen to be implemented during plant life extension refurbishment:

- Propose new additional trips to the safety systems as well as the modification of some of the existing trips which are required to improve the defence in depth against the accidents already covered by the current trips. These proposals were completed.
- Implement the trips and modifications above mentioned. These improvements *were* implemented by the end of *2018.*

PARs installation

PARs installation required for the hydrogen management inside the containment *was* implemented during the life extension refurbishment.

Filtered Containment Venting system

A filtered containment venting system *was* installed *during Refurbishment Outage*.

ECCS reliability

Improvements in the ECCS reliability consisting in changes to:

- Guarantee the injection.
- Increase the system reliability operation.
- Avoid coolant leaks to the ECCS (containment by-pass).

These improvements *were* implemented during plant life extension refurbishment.

Implementation of a rupture disc on inspection port of calandria vault

During CNE Life Extension, a rupture disc *was* added on inspection port of the calandria vault in order to allow the reduction of its internal pressure in case of severe accident.

Facility to connect a fire-truck from outside the spent fuels storage pool building

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 was installed.

Instrumentation and Control

The I&C located in the reactor core and the containment necessary to dispose of the required information for the severe accident management were installed during plant life extension refurbishment.

3.14.3.1.3.4. Others Strategies proposal

Progress in the implementation of some additional strategies to those proposed from Fukushima analysis, is detailed below:

CNA I

- Small LOCA strategy, the basic engineering design changes needed to implement the primary system inventory refilling strategy by means of the volume and pressurizing control system (TA) was completed. The corresponding guideline has been drawn up. *(Guides 1-GAS-CE-07 Revision 3, 1-GAS-SC-07-1 Revision 3 and 1-GAS-SC-07-2 Revision 3).*
- The engineering tasks in order to implement the recovery of the second heat sink system (SHS) capability by injecting demineralized water to SHS make up tank were analysed. The corresponding

guideline has been drawn up. *(Guides 1-GAS-CE-05 Rev. 4, 1-GAS-SC-05-1 Rev. 3, and 1-GAS-SC-05-2 Rev. 3).*

- *Guide 1-GAS-SC-05-3 Revision 1, corresponding to the water injection to the steam generators by pressurizing the feed water tank, was prepared.*
- A report was issued regarding the potential strategies of the containment management during a severe accident. Different containment venting strategies were examined and the possibility of the steel containment refrigeration by internal and external circulation was considered. *Two guides were developed for the Reduction of the pressure in the containment (1-GAS-CE-10-2 Revision 2, 1-GAS-SC-10-1 Revision 1 and 1-GAS-SC-10-2 Revision 1).*
- *Instruction T-42 "Unavailability of the Main Control Room" will be incorporated into the Operations Manual, which will enable RESA to be made from the Emergency Control Room located in the building of the Second Heat Sink, and bring the plant to a safe condition, until the Main Control Room can be recovered. This instruction is in process of approval.*

Potential effects on other nearby plants

Potential effects of CNA I over CNA II and vice versa were analysed and emergency plan was modified taking into account this effects.

CNA II

Accident management measures currently available to deal with the successive stages of a scenario of cooling function loss of the fuel storage pools: It is extremely unlikely that fuel element degradation could be reached, because there would be 188.5 hours to take actions to recover the cooling system. Taking into account that the worst case scenario (pool filled with all the fuels corresponding to a reactor core) is very unlikely to occur, if the spent fuel storage 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.

CNE

As it was mentioned in previous CSN Reports the main challenges identified are due to hydrogen production slow the over-pressurization of the containment and interaction of the molten core with concrete of the calandria enclosure (MCCI). Different guidelines had been developed and some of them are now under review process.

3.14.3.1.3.5. Development of Instructions and Guidelines

The Severe Accident Management Manual was developed (applicable to Units I and II), which gives structure and mechanisms to develop the Severe Accident Management Program and main guidelines for the management of an accident situation.

Procedures have been developed for Units I and II, regarding the organization of the program, generation / control / filing / distribution of documentation and their technical adequacy.

Several guidelines have been developed (*called GAS and GDC* guides) within this program, guidelines for the Emergency Command (coded as CE) and guidelines for the Control Room (coded as SC). In case they are linked, they are given the same number.

GAS Guidelines include actions to be taken when core integrity and / or containment are compromised. The negative impacts of each strategy must be carefully analyzed.

GDC Guidelines includes actions to be taken when the integrity of containment is compromised. They do not take into account the negative impacts that may result from their execution. They have higher setpoints.

So far the following *GAS* guidelines for CNA I were developed:

- *Guideline CNA I-II AG 01 (Rev. 2): Rules to use guidelines A, applicable to both Units.*
- *Guideline 1-GAS-CE-01 (Rev. 3): Evaluation of plant status.*
- *Guideline 1-GAS-SC-01 (Rev. 3): Main Guidelines for Control Room.*
- *Guideline 1-GFAS-CE-01 (Rev. 3): Long-term monitoring.*
- *Guideline 1-GFAS-CE-02 Guide (Rev. 2): Completion of Guidelines A.*

- *Guideline 1-GAS-CE-04 (Rev. 2): Power failure.*
- *Guideline 1-GAS-SC-04-01 (Rev. 2): Interconnection from Unit II to Unit I.*
- *Guideline 1-GAS-SC-04-02 (Rev. 2): Mobile Diesel Generator.*
- *Guideline 1-GAS-CE-05 (Rev. 4): Feed and Bleed of Steam Generators.*
- *Guideline 1-GAS-SC-05-01 (Rev.3): Water Injection to the Steam Generators - High Pressure Way.*
- *Guideline 1-GAS-SC-05-02 (Rev.3): Water Injection to the Steam Generators - Low Pressure Way.*
- *Guideline 1-GAS-SC-05-03 (Rev.1): Water Injection to the Steam Generators by Pressurizing the Feed Water Tank.*
- *Guideline 1-GAS-CE-07 (Rev.3): Water Injection to the Primary.*
- *Guideline 1-GAS-SC-07-01 (Rev.3): Water Injection to the Primary (TA).*
- *Guideline 1-GAS-SC-07-02 (Rev.3): Water Injection to the Primary (TA/TN).*
- *Guideline 1-GAS-CE-08 (Rev.1): Water Injection to Containment Sinks.*
- *Guideline 1-GAS-CE-09 (Rev.2): Reduction in the release of fission products.*
- *Guideline 1-GAS-SC-09-01 (Rev.1): Insulation of Containment.*
- *Guideline 1-GAS-CE 10 (Rev. 2): Control of Containment Conditions.*
- *Guideline 1-GAS-SC-10-1 (Rev. 1): Containment Relief (TL7).*
- *Guideline 1-GAS-SC-10-2 (Rev. 1): Containment Relief (TL8).*
- *Guideline 1-GAS-CE 12 (Rev. 4): Water Injection to the Spent Fuel Elements Pools.*
- *Guideline 1-GAS-SC-12-1 (Rev. 3): Water Injection to the Spent Fuel Elements Pools (UJ04).*
- *Guideline 1-GAS-CE 13 (Rev. 2): Refrigeration of Loading Machine.*
- *Guideline 1-GAS-SC-13-1 (Rev. 1): Refrigeration of Loading Machine (SBO).*

(GDC) Challenges to Containment Guidelines are incorporated into the management of severe accidents.

- *1-GDC-CE-01 Guide "Evaluation of Containment Challenge", Revision 0.*
- *1-GDC-CE-02 Guide "Reducing the release of fission products", Revision 0.*
- *1-GDC-CE-03 Guide "Reduce the pressure within the containment", Revision 0.*
- *1-GDC-CE-05 Guide "Containment vacuum control", Revision 0.*

So far the following *GAS* guidelines *were developed* for CNA II NPP*:*

- *Guideline CNA I-II AG 01 (Rev. 2): Rules to use guidelines A, applicable to both Units.*
- *Guideline 2-GAS-CE-01 (Rev. 4): Evaluation of plant status.*
- *Guideline 2-GAS-SC-01 (Rev. 3): Main Guidelines for Control Room.*
- *Guideline 2-GFAS-CE-01 (Rev. 3): Long-term monitoring.*
- *Guideline 2-GFAS-CE-02 Guide (Rev. 1): Completion of Guidelines A.*
- *Guideline 2-GAS-CE-04 (Rev. 3): Power failure.*
- *Guideline 2-GAS-SC-04-2 (Rev. 1): Plant Refrigeration with One Electric Train Active.*
- *Guideline 2-GAS-SC-04-4 (Rev. 1): Electrical Interconnection from Unit I to Unit II.*
- *Guideline 2-GAS-SC-04-8 (Rev. 0): Feeding from Emergency DG from Unit I to Unit II.*
- *Guideline 2-GAS-CE-05 (Rev. 2): Feed and Bleed of Steam Generators.*
- *Guideline 2-GAS-SC-05-1 (Rev.2): Water Injection to Steam Generators (LAB/LAH).*
- *Guideline 2-GAS-CE-06 (Rev. 1): Depressurization of Primary.*
- *Guideline 2-GAS-CE-07 (Rev. 2): Reduction of Release of Fission Products.*
- *Guideline 2-GAS-SC-07-1 (Rev.1): Insulation of ventilation of containment.*

- *Guideline 2-GAS-CE-09 (Rev. 1): Water Injection to the Primary.*
- *Guideline 2-GAS-SC-09-1 (Rev.2): Water Injection to Primary (KBA).*
- *Guideline 2-GAS-CE-11 (Rev. 3): Water Injection to the Spent Fuel Pool.*
- *Guideline 2-GAS-SC-11-1 (Rev.3): Water Injection to the Spent Fuel Pool (GHC).*
- *Guideline 2-GAS-SC-11-2 (Rev.0): Water Injection to the Spent Fuel Pool (SG).*
- *Guideline 2-GAS-CE-13 (Rev. 1): Water Injection to Containment Sink.*
- *Guideline 2-GAS-CE-14 (Rev. 1): Containment Conditions Control.*

(GDC) Challenges to Containment Guidelines are incorporated into the management of severe accidents.

- *2-GDC-CE-01 Guide "*Evaluation of contention challenge", Revision 0.
- 2-GDC-CE-02 Guide "Reducing the release of fission products", Revision 0.
- 2-GDC-CE-03 Guide "Reduce the pressure within the containment", Revision 0.
- 2-GDC-CE-05 Guide "Containment *vacuum control", Revision 0.*

The following internal instructions were generated, executed by areas defined in the Organizational Chart of the Emergency Response Organization (ERO):

- *ORE-006: Operation of mobile diesel generator BY06D001.*
- *ORE-008: Fuel supply to emergency diesel generators CNA UI-II.*
- *ORE-016: Fuel oil from tank EGB01BB001 to weekly tanks of emergency diesel generators.*
- *ORE-015: Water supply to the tanks of the GHC system - Demineralized water from the SGA fire system- Fire network.*

The instruction I-01 "Power supply SGA system from the fire network of the UG-PN workshop" of CNA II, is in the process of being designated as internal instruction ORE-007 (under review).

It is noted that according to the different modifications to the facility to be carried out, the following guidelines will be generated or modified:

- *Guidelines: Supply of water to the enclosure of the pressure vessel. (In process, since it will be implemented after completion of the studies and corresponding modifications).*
- *Guidelines: Connecting the temporary cooling system of diesel generators.*
- *Guidelines: mobile diesel generator.*

The Severe Accident Management Guidelines (SAMG) for CNE -based on severe accident progression analysis performed with MAAP4-CANDU code - was already prepared and now is under revision. In December 2012, CANDU Energy performed a Severe Accident Management Guidance (SAMG) Training and Validation Exercise for the Severe Accident Management Program. The overall objectives of the SAMG validation exercises were to evaluate:

- The effectiveness of SAMG framework, processes and training for emergency response, and
- The adequacy of the communications between key CICE members, the Technical Support Group (TSG) Team Leader and Guideline Evaluators and the MCR staff.

The following four set of guidelines was defined:

- SACRGs: Severe Accident Control Room Guidelines.
- SAGs: Severe Accident Guidelines.
- SCGs: Severe Challenge Guidelines.
- SAEGs: Severe Accident Exit Guidelines.

A final revision of the following guides is being conducted:

- 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.

It should be mentioned that some preventive strategies were included in Operating Manuals as extension of abnormal operating procedures or instructions.

3.14.3.1.4. CAREM Deterministic Safety Assessment

In order to simulate the plant response in case of design basis events, a plant model was developed using the plant code RELAPSIM. For severe accidents, reactor and containment models were developed using MELCOR code.

The deterministic safety assessment (DSA) includes more than 40 sequences, including those with the failure of the first shutdown system (ATWS). The analysis for most of the sequences was extended to 36 hours (plat grace period).

In all cases, the acceptance criteria are fulfilled with large margins, being these acceptance criteria more restrictive than those internationally accepted for PWRs, as it is the case of LOCA where DNBR>1.25 and core covered is required.

3.14.3.2. PROBABILISTIC SAFETY ASSESSMENT

The upgraded activities covering period between 2013 and *2019* in the applications of PSA are the following:

3.14.3.2.1. CNA I Probabilistic Safety Assessment applications

The PSA L1 study of CNA I was developed from the nineties. The original scope included Internal Events at full power operation. Nevertheless, the study was continuously improved incorporating the changes that were made in the plant. Among others, the following important cases must be mentioned: Secondary Heat Sink (SHS), redundancy improvements as in the secondary relief valve system. Also the PSA L1 study was used as a tool to decide on some of the modifications.

With respect to the operation, PSA L1 is used to analyse the effects on risk related to projected changes and helping decision making. For instance, in 2005, NA-SA submitted a technical evaluation to ARN supporting a request for increase the time between planned outages, from 12 to 18 months. The impact on nuclear safety due to frequency modifications in the preventive maintenance activities, periodical test implementation and ISI programs was evaluated. The augmented test period was the main subject considered and an estimation of the impact on the CDF using PSA was carried out, which turned out to be $1.3x10^{-5}$/year. This result combined with other assessments was the basis for rejecting this modification of the above procedures.

The PSA L1 Rev. 3 study was completed by the Licensee in 2009 and submitted to the Regulatory Body for review and approval. This PSA L1 Rev. 3 has included an update of the significant design modifications and incorporated results from new deterministic studies conducted in the framework of SAMP for CNA I. The scope included full power internal events.

These studies extended the model of the main sequences by using the RELAP code until the occurrence of the core discovery leading to its damage which allowed removing conservative assumptions used in the previous reviews. These new simulations with RELAP allowed a more realistic quantification of some accidental sequences which demand the SHS system action. As an example, could be mentioned the occurrence of a small LOCA through the pressurizer safety valve coincident with the failure of the low pressure emergency core cooling system injection which, in previous reviews, led to core damage and, with the new studies this scenario not leads to core damage along the calculated mission time.

The Regulatory Body finished the review of the above mentioned PSA and presented the review results to the Licensee including the corresponding findings. As a consequence, it was agreed to perform a further PSA L1 updated, which has been completed in 2015. Among the improvements required by the Regulatory Body, the use of a cut criterion considering the involved frequencies for the sequences contributing to CDF to ensure the results stability could be mentioned. Moreover, deterministic assessments to justify the success criteria adopted in some accidental sequences for the heat removal systems were required.

The PSA L1 Rev. 3 has also been used to address a seismic margin study, by analysing the sequences and components of greater contribution to risk. The methodology chosen was the seismic margin assessment (SMA) -EPRI methodology-, and the PSA was used as the main tool to identify the components to be included in the equipment list required for the plant safe shutdown (SSEL, Safe Shutdown Equipment List).

In addition, the Regulatory Body has reviewed the PSA as one of the safety factors of the plant periodic safety review, resulting in the identification of model improvement opportunities as well as that expanding the scope of the study also is needed. The methodology used for this review was based on comparing the PSA content with the international guidelines and recommendations applicable, such as the IAEA Guide SSG-3 "Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants ".

The PSA L1 Rev. 4 study corresponding to full power internal events was completed by the Licensee in 2015 and submitted to the Regulatory Body. The following design changes have been included in this updated version of PSA L1 model:

- New system for emergency power supply.
- Installing the fourth UK pump.
- Changes in pumps interlocks of the moderator.
- Manual power supply interconnection between Units I-II.

In addition to the updating with new mentioned characteristics of the plant some conservative assumptions of plant behaviour were avoided. Also some model verification and improvement were considered. Among others, the following should be mentioned:

1. Incorporation of the findings of new deterministic studies: Deterministic studies have confirmed that the grouping of small LOCA S2AA included in the previous review is correct.
2. Incorporation of the operating experience over the years from previous PSA Revision 3: the statistics of initiating events has been updated with information corresponding to period between 2008 and 2013.
3. As a result of the modifications review and new modelling of safety systems were considered: new grouping of initiating events, new outlines of fault tree events related to power supply, new models of fault trees related to power supply and incorporation of new headers, among others.

It is also noted that this review is including the recommendations made by the Nuclear Regulatory Authority regarding revision 3, such as changing the cut-off values for quantification.

Review 1 of the Internal Fire Analysis (Phase 2 APS Level 1) is being updated by incorporating the Second Heat Sink, the new Emergency Power Supply System and the fourth UK pump. The following stages have already been met:

- Review of existing fire zones and identification of new fire areas resulting from the modifications to the facility already mentioned.
- Definition of initiating events.

The application of PSA results in the safety assessments performed as a response action after Fukushima is the following:

- In order to check the availability of the SSCs which are required in accident scenarios, a SSCs list to check on the walk-downs recommended by WANO in 2011 was elaborated from the PSA. There were passive components which were added to this list that had been not modelled in the PSA. However, they were considered, according to expert judgment and based on its importance for the plant response to accidental situations.
- To comply with Recommendation 3 of WANO SOER 2011-2, it has been studied the availability of the SSCs that must be functional in case of external flooding scenarios. The SSCs necessary to stop, bring and maintain the reactor into a safe shutdown condition after a loss of the river

water assured cooling system (UK), were identified from the plant's PSA as well as their availability was verified through the plant walk-downs.

- The PSA was used as the main tool to identify the components to be included in the SSEL.

Currently the revision 5 of the Probabilistic Safety Assessment Level 1 for internal events is approved.

This revision 5 *included* in the plant model the changes introduced due to:

- Updating the frequency of low frequency initiating events or events that have never happened considering not only own experience of the plant, but also based on a search of external operating experience and the methodology currently used in the bibliography. Allocating the corresponding data uncertainties.
- Updating the frequency of initiating events that have an associated fault tree for their quantification. Allocating the corresponding data uncertainties. Directly associate the modelled fault tree to the initiating event of the corresponding fault tree.
- Revision of the frequency of initiating events that have associated plant shutdowns corresponding to each initiating event by applying the Bayes Theorem, that is, taking into account own operating experience. Allocation of corresponding data uncertainties.
- Update probabilities or frequencies of failures and unavailability of basic events of CNA I APS considering generic database and own operating experience from different sources such as: records of repetitive tests, the database of internal events and history of plant by Visual Data program. Allocation of corresponding data uncertainties.
- Categorization of basic events of CNA I APS in families of components.

In addition, other general improvements were incorporated, eliminating conservative assumptions in the model behaviour of the plant safety systems which included: a review of the criteria for success and modelling of certain headers, general improvements to the modelling of emergency electrical system that led to modify common cause failures and time of testing of basic events, incorporating common cause failures replacement of basic events by an existing or new fault tree or, improved modelling of power supply from several bars and general corrections to the modelling of the safety systems by default or incorrect coding, among others.

Review 1 of the Internal Fire Analysis (Phase 2 APS Level 1 *was developed* by incorporating the Second Heat Sink, the new Emergency Power Supply System and the fourth UK pump, and *based on PSA Revision 5. It was submitted to the Regulatory Body.*

Currently, Revision 1 of the Probabilistic Safety Assessment Level 1 is being carried out for other operational states. The purpose of this revision is to:

- *Calculate the Core Damage Frecuency in other operating states other than full power operation.*
- *The PSA model must be able to represent the Plant in any stop state and the unavailability status of the equipment.*
- *The PSA model must allow calculate the risk and represent the plant during the planning and execution of any stop plant state.*

Also it has to include, in the analysis of other operating states, the changes introduced by:

- *The incorporation of the Second Heat Sink (SHS).*
- *New Emergency Power Supply (EPS).*
- *Installation of the fourth pump in the UK system.*
- *Updating of the database contemplated in Revision 5 of the PSA of the CNA I.*

3.14.3.2.2. CNA II Probabilistic Safety Assessment applications

For commissioning of CNA II it was required to develop PSA Levels 1 to 3. The objective was to demonstrate that the probabilistic standard of the Argentinean regulation are fulfilled (Regulatory Standard AR 3.1.3., Radiological Criteria Relating to Accidents in Nuclear Power Plants). Basically this norm establishes that the product of frequencies and consequences for severe accident sequences have to be below certain threshold.

In the period 2009-2012, the Licensee completed the first PSA L1 revision corresponding to internal events at full power. This version was reviewed by the Regulatory Body in 2011 and, as a result, the following main requirements and recommendations were carried out:

- *New deterministic studies including LOCAs affecting the moderator system must be provided.*
- *New initiating events not included in the original version such as high pressure PHTS transients, LOCA's BDBA and loss of three secured service water system trains must be added to the study.*
- *Re-evaluations to update the frequencies of some initiating events are needed: loss of external electrical power networks, loss of service water supply by using the CNA I operational experience.*
- *Updating of some component failure rates / probabilities considering the accumulated operational experience of CNA I.*
- *Inclusion of the loss of main heat sink event tree (in the original version this case was included in the generic transient event tree).*

The updated CDF value obtained from the PSA L1 model for internal events at full power is 1.9 10^{-5}/year.

The Regulatory Body issued additional requirements for the PSA L1 to be met for the issuance of the Operating License of the plant:

- *To complete the preliminary human reliability analysis (HRA) performed for pre-initiator human actions, incorporating the results in the PSA L1.*
- *To complete the preliminary human reliability analysis performed for post-initiator human actions (category C1), focusing analysis on those accident sequences which have the greater impact on the core damage frequency (e.g.: Station blackout) and considering the contribution of associated "hardware" failures. Results have to be incorporated in the PSA L1.*

The Licensee performed a systematic human reliability analysis for pre-initiator human actions, comprising maintenance, calibration and test tasks applying NUREG/CR-4772. For calibration tasks, the study and results were delivered to the Regulatory Body in 2014; the analysis has to be completed incorporating plant experience. For test and maintenance tasks, the study and results were delivered to the Regulatory Body in 2015.

Similarly, Licensee performed a human reliability analysis for post-initiator human actions applying NUREG/CR-4772. The studies and results were delivered to the Regulatory Body in 2016.

The HRA studies were reviewed by the Regulatory Body, considering that the Licensee had basically fulfilled the PSA requirements for Operation License Issuance (May 2016).

Requirements related to the extension of studies previously performed were issued by the Regulatory Body. These requirements have to be developed during the operating phase of the plant and included in the next version of PSA L1:

- *To extend the HRA for all calibration tasks included in the correspondent Program.*
- *To extend the HRA for post-initiators human actions to include those that will arise from the implementation of plant modifications related to post-Fukushima issues.*

Additionally, during this period (2013-2019) the following PSA L1 tasks were initiated and are currently in progress:

- *Updating of reliability parameters components data to include the applicable ZEBD Database.*
- *Updating of reliability parameters components data to include specific plant operating data (preventive maintenance frequencies, components repair times, definitive components test schedules, etc.).*
- *Preparation of the PSA L1 model for specific applications. Particularly, verification of AOT using probabilistic techniques.*

Besides, in parallel to the PSA L1 the Licensee has developed the corresponding PSA L2 and L3. The PSA L2 analyses accident progression of the sequences identified by the L1-L2 interface. The accident sequence grouping was based on the identification of a set of attributes that characterize similar states.

The PSA L2 was delivered in 2011 and a new version (which was updated considering the ARN comments) was issued in April 2013.

The PSA L3 developed uses the results of the PSA L2 as the starting point. The PSA L2 results consists in the frequency and source term of each one of the release categories defined in the L2-L3 interface. The source term is characterized by the amount, type and energy of radioactive material and the containment failure path. The PSA L3 assesses the atmospheric dispersion as well as the resulting dose to the public. Finally, the maximum radiological risk to the public corresponding to each release category is calculated to verify the fulfilment of the criterion established by the Standard AR 3.1.3. - Radiological Criteria Relating to Accidents in Nuclear Power Plants.

The PSA level 3 was delivered in August 2011 and a new version (which was updated considering the ARN comments) was sent to the ARN in April 2013.

The review of PSA L2 and L3 was performed by the Regulatory Body with the assistance of Sandia N.L. and GRS during 2015.

3.14.3.2.3. CNE Probabilistic Safety Assessment applications

In 2004 the Licensee completed a first PSA L1 version considering full power internal events. This version has included corrective actions. Later on, the PSA scope was expanded including low power and shutdown operational conditions and other potential radioactive sources (spent fuel storage pool, cobalt rods, dry spent fuel storage, etc.).

Regarding applications, in some cases, PSA was used to demonstrate the suitability of any proposed modification intended to justify and obtain approval from the Nuclear Regulatory Authority to implement change.

In another example of application, PSA study was used to assess the effect on risk due to higher testing period from 12 to 18. As part of the decision making process, the ARN required to evaluate the impact on the nuclear safety due to such increase in the time between application programs of preventive maintenance, periodical tests and in service inspection.

Another field of application is the use of the models of the systems developed for the PSA in the calculation of reliability indicators of certain relevant systems. The indicator is obtained by calculating of the failure probability of a given system, with certain additional conditions to the PSA model, and considering the real past time in which the components of the system were out of service for maintenance. This result is compared with an established target value for every system.

The CNE NPP has finished first operation cycle at the end of 2015 and started refurbishment outage *in 2016*. The Licensee has already developed a new PSA L1 revision to update the previous one considering scope of full power internal events and including the *plant configuration that will face the second period of operation*. This new PSA L1 version *was* finished during 2015 *and it was an input for the regulatory decision to approve the proposed safety system modifications*.

In addition to these safety system modifications, the Licensee has considered new deterministic results corresponding to transients and postulated accidents that allow reviewed some accidental sequences. The results have ruled out some non-conservative hypotheses included in the previous PSA L1 version, such as the use of the moderator as last heat sink in certain LOCA cases. Regarding this, although until now it had been considered an effective mechanism for heat extraction under these conditions, according to the new available results, this cannot be considered as credible. Results obtained with the new version of the internal events PSA indicated a CDF value of about *3.46* $\times 10^{-5}$ / year.

Another important modification introduced in the PSA review is related with core damage states CDS grouping by adapting these states to the definition that requires the continuation of L1 to L2. The results and experience in similar CANDU plants are used in PSA L2 study, grouping the damage categories according to the characteristics of the consequences to be had in each case. For each damage states contributors' sequences coming from different internal initiating events are grouped. The CNE PSA L2 has been finished during 2015.

The plant seismic safety assessment has been performed by using a SMA *(Seismic Margin Assessment)* based on PSA, identifying the PSA sequences of internal events with the corresponding plant final states which cause severe core damage. Therefore, based on the above, the SSCs *(Structure, System and Component)* to be included in the SSEL *(Safe Shutdown Equipment List)* have been identified.

3.14.3.2.4. CAREM Probabilistic Safety Assessment applications

In 2010, a PSA L1 within the scope of the basic engineering stage was developed. It was carried out using the technique of large event trees, including, as headings active and passive systems involved in the control of the initiating events. The Risk Spectrum code was used to solve both the fault trees and the event trees.

A PSA L2 was developed based on expert judgment, NUREG 1150 and assuming very conservative hypothesis.

A PSA L3 was developed where the individual radiological risk in the public was quantified and the regulatory acceptance criterion AR 3.1.3. was fulfilled.

3.14.3.3. PERIODIC SAFETY REVIEW

Since 2003 ARN has changed the methodology for license renewal of NPPs in operation, including Periodic Safety Review (PSR) as a licensing tool.

The requirement to develop PSR was included in the operating license of CNA I, CNA II and CNE. It was established that PSR have to be developed each 10 years, with the scope described in the IAEA Safety Standard Series SSG-25 "Periodic Safety Review of Nuclear Power Plants" document. Approval of PSR results by ARN is a necessary condition for license renewal.

CNA I presented his first PSR in 2013. The safety factors for CNA I PSR are listed below.

1. Plant design.
2. Actual condition of SSCs important to safety.
3. Equipment qualification.
4. Ageing management.
5. Deterministic safety analysis.
6. Probabilistic safety analysis.
7. Safety performance.
8. Use of operating experience.
9. Organization, the management system and safety culture.
10. Procedures.
11. Human factors.
12. Emergency planning.
13. Radiological impact.

As it was mentioned in this National Report, CNA I decide to proceed with the long term operation until 56.2 full power year (see Section 3.6. for more information). In order to develop a program for the project, the PSR had to be enlarged specifically in the plant safety factors in order to consider an appropriated scope dealing with one time efforts like Time Limited Ageing Analysis (TLAAs), Ageing Management Review in addition to the regulatory expectations for continued operation beyond the design lifetime.

Although the plant had already implemented important design changes to upgrade the original design (CNA I was designed in the 60´s), the results of the review of safety factor "Design" against modern standards including the German KTA rules, identified new improvement areas like equipment qualification, more severe internal flooding protection, segregation and separation of safety systems, etc.

In the case of the CNE due to the fact that it is engaged in a plant life extension programme, PSR is being developed as part of the safety assessments for that project. However, aging evaluations were completed and design improvements that were introduced during refurbishment outage emerged from the results of safety evaluation made in others CANDU plants.

The results of comparison with international standards, in particular Canadian design standards gave an important input to the design changes considered in the plant refurbishment.

The participation of CANDU Energy in the comparison with Canadian standards, which were included in the scope of several safety factors, should be mentioned as strength of the review.

As a condition for renewal of the operating license for the second cycle, ARN requested a PSR with a comprehensive scope which includes an important number of standards for each safety factor. Its regulatory review is an on-going activity.

The safety factors of CNE PSR were identified as follows:

1. Plant design.
2. Actual condition of SSCs important to safety.
3. Equipment qualification.
4. Ageing.
5. Deterministic safety analysis.
6. Probabilistic safety analysis.
7. Hazard analysis.
8. Safety performance.
9. Use of experience from other plants and research findings.
10. Organization, the management system and safety culture.
11. Procedures.
12. Human factors.
13. Emergency planning.
14. Radiological impact.
15. Severe Accident Management.

3.14.4. REGULATORY PLANT SAFETY PERFORMANCE

Since 1998, safety performance indicators (SPIs) data was collected and evaluated as was explained in previous national convention reports. Until 2002, the SPIs evaluation was made throughout the analysis of changes in their trends, but there wasn't any acceptability criterion.

In order to establish thresholds or acceptability values for SPIs, it was necessary to analyze historical data, but getting historical data was not possible for most of the SPIs at the beginning of the program. Statistics was made for those SPIs that were reported in the past (outages, power reductions, dose, training, wastes and effluents), but such a method was not applicable for SPIs in areas like maintenance or repetitive tests.

In 2002, frequency distributions of each SPI were made using the data collected since 1998. From those distributions, an acceptability criterion was defined and a pilot implementation was initiated for validation. As a result of the pilot implementation experience, evaluation criteria were changed.

Thresholds for SPIs of each NPP were calculated separately because plant performances are not comparable.

From 2006 and taking into account its own experience plus IAEA documentation, the number of SPIs used in ARN was modified to 24. The set in current use is well detailed in the previous CNS Report, covering different aspects of normal and abnormal operation conditions.

It should be mentioned that Argentine experience in the use of SPIs show that they by themselves are not sufficient to assess safety during NPP operation. However they contribute to have an estimation of the safety status and tendencies. They represent a useful tool to plan inspections, audits and some special regulatory assessments.

Furthermore, this set of SPIs is used as a regulatory tool to provide an additional view of the NPPs performance, allowing improving the ability to detect any eventual degradation on safety related areas. It is a satisfactory tool for monitoring safety but not using it on its own but together with other tools, such as event analysis, audits and inspections.

If the SPIs values that are quarterly reported or their trend over some period shows potential safety degradation, it is analysed in the context of the rest of the SPIs and the plant conditions (events, unplanned outages, particular conditions of operation, etc.). Besides, more information coming mainly from inspections, audits and assessments (including the analysis of the direct cause and the root cause) is collected in order to confirm the diagnosis. Also, corrective measures taken by the licensee are assessed to decide the corresponding regulatory actions to be taken.

The validation or modification of the defined limits is a continuous task. In particular, during 2012, it was performed an exhaustive revision of the SPIs values over the 1998-2011 period included on the statistics in order to validate the limits and thresholds of the SPIs.

Since January 2016 SPIs for CNE were reduced to 15 considering the change of the operation state of the plant during refurbishment outage.

Reporting of the same set of SPIs was included as a requisite in the Operating License of CNA II.

3.14.5. COMPLIANCE WITH THE OBLIGATIONS IMPOSED BY THE CONVENTION

The information contained in this section related to probabilistic and deterministic safety assessment of Argentine NPPs in operation, demonstrates that the country complies with the obligation imposed in Article 14 of the Convention of Nuclear Safety.

3.15 ARTICLE 15: RADIOLOGICAL PROTECTION

Each Contracting Party shall take the appropriate steps to ensure that in all operational states the radiation exposure to the workers and the public caused by a nuclear installation shall be kept as low as reasonably achievable and that no individual shall be exposed to radiation doses which exceed prescribed national dose limits.

3.15.1 INTRODUCTION

The control and supervision of NPPs compliance with regulatory standards and other regulatory documents are carried out over a program of routine and non-routine inspections and audits.

This program is carried out by resident inspectors and other technical support (TS) groups belonging to the ARN, who perform independent analyses and assessments related to radiological safety. These TS groups have capabilities to perform the measurements and evaluations required for this purpose, and also their own laboratories.

Although these control tasks are routinely performed, they are carried out especially in some specific situations, such as outages and unforeseen shutdowns.

The periodic test programs related to radiological protection performed by the licensee of the NPPs are verified by ARN. This includes radiation detection equipment installed in controlled and supervised areas and personnel dosimetry system. Also, licensee´s dosimetry laboratories participate of intercomparison exercises organized by ARN.

Authorized values to the environmental releases of the NPPs were set by the Regulatory Body for relevant radionuclides. They were calculated considering the dose, due to all exposure pathways, for a hypothetical representative person.

The NPPs licensees routinely measure the radioactive releases and reports the corresponding values as required. ARN has an auditing program applied to the release measurement procedures and it carries out benchmarking exercises on pattern measurements, procedure control, measurement devices and calibration. Furthermore, the Licensee and the Regulatory Body perform independent measurements of activity concentration on environmental samples and public dose evaluation using environmental models that consider “concentration factor methods”, used by UNSCEAR and recommended in the IAEA Safety Reports Series 19 and Safety Series 57. Each model has specific plant information such as the location of representative person, habits and food consumption and local dispersion factors of environmental releases. Also by keeping a historical database of the results from the environmental monitoring, an extra control of the releases can be done by comparing each new environmental sample against its corresponding historical value.

ARN requests the licensee to carry out an environmental monitoring plan, which must be reviewed and approved by the regulator. The monitoring plan objectives are primarily to verify the impact of the discharges on the environment, checking the compliance with dose limits to the public. The plan implies the sampling of environmental media such as surface and underground water, sediments, air, soil and foodstuffs, mainly drinking water, locally grown vegetables and locally produced milk. Samples are analysed for Tritium, total alpha and beta emitters, some transuranic elements, Strontium 90, Cobalt 60, Cesium 137 and Iodine 131. Sampling points are selected according to models of radionuclide dispersion in the environment, considering atmospheric and liquid discharges separately. At least three main sampling points are required: a Background point (upwind and upstream), a Maximum Expected Concentration point (downwind and downstream), and whenever possible, a point covering the Representative Person (Real). ARN performs its own independent radiological environmental monitoring plan with lower sampling frequency, using its own specialists and laboratories, covering the same environmental media and radionuclide analysis, as well as the same required sampling points.

3.15.2. CNA I

3.15.2.1. RADIOACTIVE RELEASES TO THE ENVIRONMENT

The estimation of most relevant radioactive releases to the environment due to CNA I operation in the *2016-2018* period are presented in Table 3.15.1. and Table 3.15.2.

Table 3.15.1. - Activity released from CNA I to the environment as gaseous discharges

YEAR	I-131 (TBq)	TRITIUM (TBq)	AEROSOLS (TBq)	NOBLE GASES (TBq)
2016	*3.4×10^{-3}*	*5.0×10^{2}*	*1.7×10^{-6}*	*9.7 x 10*
2017	*3.7×10^{-4}*	*5.3×10^{2}*	*1.6×10^{-6}*	*1.5 x 10*
2018	*8.7×10^{-6}*	*4.9×10^{2}*	*1.0×10^{-6}*	*3.6 x 10*

Table 3.15.2. - Activity released from CNA I to the environment as liquid discharges

YEAR	TRITIUM (TBq)	OTHER RADIONUCLIDES (TBq)
2016	*1.3×10^{3}*	*7.9×10^{-2}*
2017	*5.9×10^{2}*	*8.2×10^{-2}*
2018	*1.3×10^{3}*	*1.1×10^{-1}*

Of the total annual average discharges from CNA I to the environment, almost *97%* corresponds to tritium.

3.15.2.2. PUBLIC EXPOSURE

A comparison of activity concentration results from environmental samples against different derived guideline or reference levels is performed. No guideline levels were exceeded in the period from 2016 *to 2018.* The results from the licensee´s monitoring plan are presented quarterly to the ARN, which compares them to its own environmental results. No major discrepancies were found.

Tritium was detected in downwind moisture condensate samples as well as in some downstream samples (Paraná River). Also in locally produced vegetable and milk. In all cases values observed were not relevant from a public exposure viewpoint.

Cesium 137 was occasionally detected in sediment and soil samples alternatively in upwind/upstream and downwind/downstream sampling points, as well as in fish. Values measured also suggest their relation to fallout levels. Cobalt 60 was once detected in downstream sediments. All these results, however, were barely above gamma measurement detection limits for environmental samples, and are dosimetrically irrelevant.

The effective dose has been calculated for the most exposed individual considering very conservative hypothesis. The annual dose estimated values are shown in Table 3.15.3.

Table 3.15.3. - Representative person dose for CNA I

YEAR	GASEOUS DISCHARGE DOSE (mSv)	LIQUID DISCHARGE DOSE (mSv)	TOTAL DOSE (mSv)
2016	*7.0×10^{-3}*	*1.8×10^{-4}*	*7.2×10^{-3}*
2017	*7.5×10^{-3}*	*2.4×10^{-4}*	*7.8×10^{-3}*
2018	*6.9×10^{-3}*	*1.8×10^{-4}*	*7.1×10^{-3}*

3.15.3. CNA II

3.15.3.1. RADIOACTIVE RELEASES INTO THE ENVIRONMENT

The estimation of most relevant radioactive releases to the environment due to *operation of CNA II* in the *2016-2018* period are presented in Table 3.15.4. and Table 3.15.5.

Table 3.15.4. - Activity released from CNA II to the environment as gaseous discharges

YEAR	I-131 (TBq)	TRITIUM (TBq)	AEROSOLS (TBq)	NOBLE GASES (TBq)	C-14 (TBq)
2016	*1.9×10^{-3}*	*1.9×10^{2}*	*1.2×10^{-5}*	*4.7×10^{1}*	*6.1×10^{-2}*
2017	*2.7×10^{-3}*	*3.8×10^{2}*	*1.8×10^{-5}*	*3.8×10^{1}*	*2.9×10^{-1}*
2018	*7.0×10^{-4}*	*4.2×10^{2}*	*1.3×10^{-5}*	*5.2×10^{1}*	*4.3×10^{-1}*

Table 3.15.5. - Activity released from CNA II to the environment as liquid discharges

YEAR	TRITIUM (TBq)	OTHER RADIONUCLIDES (TBq)
2016	*1.9×10^{2}*	*6.3×10^{-2}*
2017	*5.9×10^{2}*	*8.2×10^{-2}*
2018	*3.1×10^{2}*	*1.3×10^{-1}*

Of the total annual discharges from CNA II to the environment, around *94%* corresponds to tritium.

3.15.3.2. PUBLIC EXPOSURE

The radiological environmental monitoring plans performed independently by the licensee and ARN, cover both CNA I and CNA II NPPs. Comments made in the CNA I section (3.15.2) are valid for CNA II.

The effective dose has been calculated for the most exposed individual considering very conservative hypothesis. The annual dose estimated values are shown in Table 3.15.6.

Table 3.15.6. - Representative person dose for CNA II

YEAR	GASEOUS DISCHARGE DOSE (mSv)	LIQUID DISCHARGE DOSE (mSv)	TOTAL DOSE (mSv)
2016	*5.5×10^{-3}*	*2.6×10^{-5}*	*5.5×10^{-3}*
2017	*4.2×10^{-3}*	*7.9×10^{-5}*	*4.3×10^{-3}*
2018	*4.6×10^{-3}*	*$4.1 \text{x} 10^{-5}$*	*4.6×10^{-3}*

3.15.4. CNE

3.15.4.1. RADIOACTIVE RELEASES INTO THE ENVIRONMENT

The estimation of most relevant radioactive releases by CNE to the environment, for the *2016-2018* period may be seen in Table 3.15.7. and Table 3.15.8.

Table 3.15.7.- Activity released from CNE to the environment as gaseous discharges

YEAR	I-131 (TBq)	TRITIUM (TBq)	AEROSOLS (TBq)	NOBLE GASES (TBq)	C-14 (TBq)
2016	*4.2×10^{-12}*	*1.7×10^{2}*	*10.0×10^{-11}*	*0.0×10^{0}*	*8.8×10^{-1}*
2017	*2.1×10^{-12}*	*1.3×10^{2}*	*8.6×10^{-11}*	*0.0×10^{0}*	*2.3×10^{-1}*
2018	*2.3×10^{-12}*	*2.6×10^{2}*	*5.2×10^{-11}*	*0.0×10^{0}*	*3.1×10^{-1}*

Table 3.15.8. - Activity released from CNE to the environment as liquid discharges

YEAR	TRITIUM (TBq)	OTHER RADIONUCLIDES (TBq)
2016	*2.3×10^{2}*	*3.7×10^{-3}*
2017	*6.3×10^{1}*	*9.5×10^{-3}*
2018	*9.4×10^{1}*	*4.0×10^{-3}*

Of the total annual discharges from CNE to the environment, around 99.98% corresponds to tritium.

3.15.4.2. PUBLIC EXPOSURE

A comparison of activity concentration results from environmental samples against different derived guideline or reference levels is performed. No guideline levels were exceeded in the period from *2016 to 2018*. The results from the licensee monitoring plan are presented quarterly to the ARN, which compares them to its own results. No major discrepancies were found. Results from 2013 monitoring indicated a rise on tritium levels in the Lake Embalse, due to a loss of heavy water from a steam generator of the plant.

Tritium was detected in downwind moisture condensate samples as well as in the Lake Embalse and downstream samples (Tercero River). Also in locally produced vegetable and milk. Tritium was also detected in drinking water, which is provided from Lake Embalse. Tritium levels in water were up to *two* order of magnitude lower than WHO guideline values.

Cesium 137 was occasionally detected in sediment and soil samples alternatively in upwind/upstream and downwind/downstream sampling points, as well as in fish. Values measured also suggest their relation to fallout levels, and resulted dosimetrically irrelevant.

The effective dose has been calculated for the most exposed individual considering very conservative hypothesis. The annual dose estimated values are shown in Table 3.15.9.

Table 3.15.9. - Representative person dose for CNE

YEAR	GASEOUS DISCHARGE DOSE (mSv)	LIQUID DISCHARGE DOSE (mSv)	TOTAL DOSE (mSv)
2016	*4.0×10^{-4}*	*6.0×10^{-3}*	*6.4×10^{-3}*
2017	*3.0×10^{-4}*	*6.0×10^{-3}*	*6.3×10^{-3}*
2018	*6.0×10^{-4}*	*4.1×10^{-3}*	*4.7×10^{-3}*

The main contributor to the representative person dose corresponds to the liquid discharges which are released into the Lake Embalse.

3.15.5. OCCUPATIONAL EXPOSURE

The radiological protection criteria used by the Regulatory Body to control the dose received by NPPs' workers are consistent with the latest ICRP recommendations.

Regulatory Standard AR 3.1.1. sets different criteria to ensure that the occupational dose to workers is as low as reasonably achievable and lower than the established dose constraints, while the protection is optimised.

ARN requires that whenever possible, radiological protection be achieved using plant's systems rather than operational procedures.

Each NPP´s Operating License sets the following conditions for their workers:

- Personnel working in a controlled area must be submitted to individual monitoring and annual medical surveillance.
- Monthly occupational doses must be recorded for:
  - External exposure.
  - Intake of radioactive material.
- These records must contain the following information:
  - Individual doses.
  - Collective effective doses resulting from the development of different maintenance, repairing and operation tasks.
- The Primary Responsible must keep the mentioned records for at least thirty years after the end of service of the involved personnel.

3.15.5.1. DOSE LIMITS TO WORKERS

The Regulatory Standard AR 10.1.1. establishes that the dose limits have not been exceeded when the following conditions are fulfilled:

$$\frac{H_p(d)}{L_{DT}} \leq 1$$

and

$$\frac{H_p(10)}{20\,\mathrm{mSV}} + \sum_j \frac{I_j}{I_{L,j}} \leq 1$$

Where:

$H_p(d)$ is the personnel equivalent dose at a depth of 0.07 mm and 3 mm (for skin and crystalline respectively), integrated in a year,

L_{DT} is the allowable limit of equivalent dose in skin or the lens of the eye,

$H_p(10)$ is the personnel equivalent dose at a depth of 10 mm from the skin surface integrated in one year,

I_j is the incorporation value of nuclide j during a year,

$I_{L,j}$ is the annual intake allowable limit for nuclide j, resulting from the division of 20 mSv by the dosimetric factor of effective dose commitment for workers, per unit incorporation of the mentioned radionuclide.

3.15.5.1.1. Occupational dose at CNA I

The collective effective dose and the average individual effective dose received by workers in CNA I during the *2016-2018* period, are presented in Table 3.15.10.

Table 3.15.10.- Occupational dose in CNA I

YEAR	COLLECTIVE EFFECTIVE DOSE (Sv.person)	AVERAGE EFFECTIVE DOSE (mSv)
2016	*3.0*	*2.4*
2017	*1.9*	*1.6*
2018	*3.0*	*2.4*

The differences in annual collective effective doses are related to the programmed outages.

3.15.5.1.2. Occupational dose at CNA II

The collective effective dose and the average individual effective dose received by workers in CNA II during years *2016 and 2018* are presented in Table 3.15.11.

Table 3.15.11.- Occupational dose in CNA II

YEAR	COLLECTIVE EFFECTIVE DOSE (Sv.person)	AVERAGE EFFECTIVE DOSE (mSv)
2016	*0.17*	*0.25*
2017	*1.16*	*0.91*
2018	*1.01*	*0.90*

3.15.5.1.3. Occupational dose at CNE

The collective effective dose and the average individual effective dose received by CNE workers during the *2016-2018* period are presented in Table 3.15.12.

Table 3.15.12. - Occupational Dose in CNE

YEAR	COLLECTIVE EFFECTIVE DOSE (Sv.person)	AVERAGE EFFECTIVE DOSE (mSv)
2016	*6.8*	*2.2*
2017	*8.7*	*2.1*
2018	*5.5*	*1.6*

The occupational doses in CNE, during *2016, 2017 and 2018* are *higher than previous years because during that period there was the Life Extension Project of Embalse NPP.*

3.15.6. ALARA ACTIVITIES

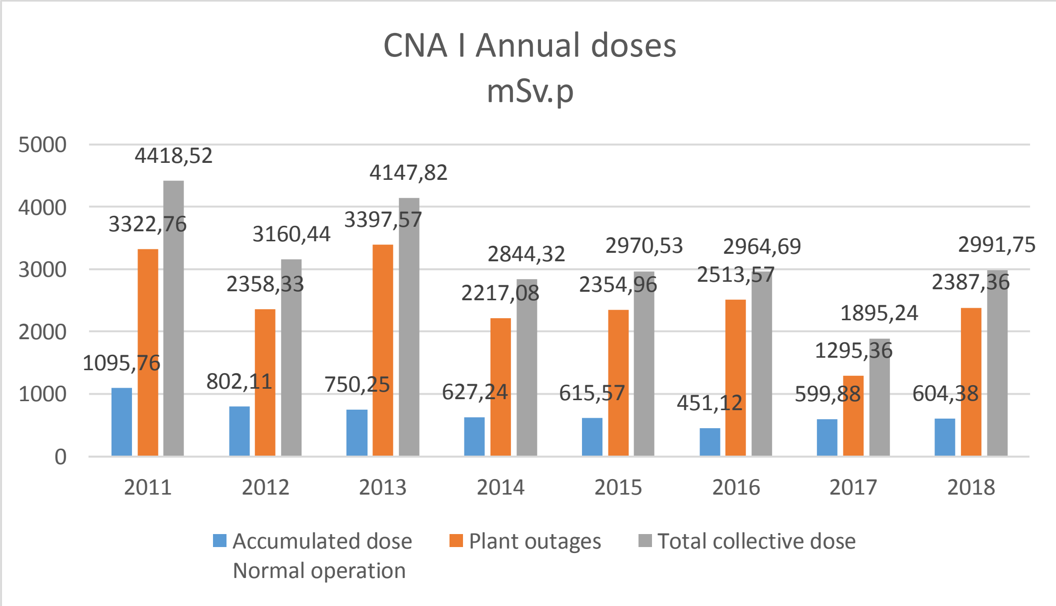
ALARA program is carried out in all NPPs during normal operation and during outages aimed to achieve constant improvement by decreasing the doses received by workers. Each NPP has a specific ALARA working group involved in the following activities:

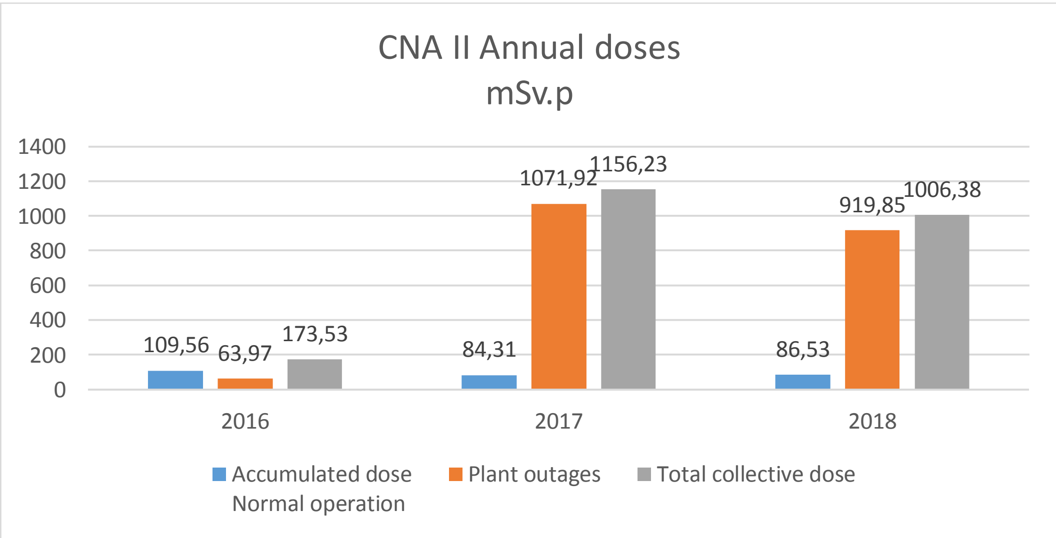
- Planning of the activities that involve doses.
- Detailed planning in case of activities that involve significant doses.
- External and internal operating experience feedback.
- Mock-up training for the above mentioned activities.
- Design of specific tools and shielding to be used in high radiation fields.

The ALARA activities have been reinforced and intensified with the objective of optimizing the dose received by personnel occupationally exposed to ionizing radiations, reinforcing its activity with better implementation of mock-ups, training, improvement in tools and the analysis of work plans.

The charts showed in 3.15.6.1. and 3.15.6.2. present some examples of dose improvements for repetitive tasks corresponding to CNA I y CNE.

Documentation and procedures for CNA I and CNA II are unified to apply the same methodology in relation to dose optimization criteria. It should be noted that the first planned outage at CNA II was in 2017.





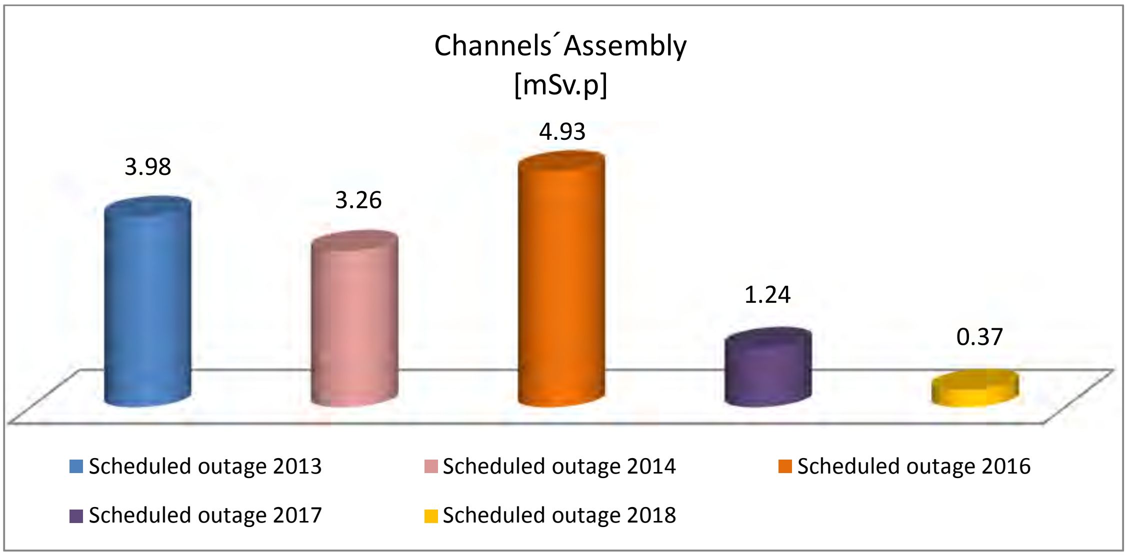
3.15.6.1. CNA I REACTOR INSPECTION PROGRAM

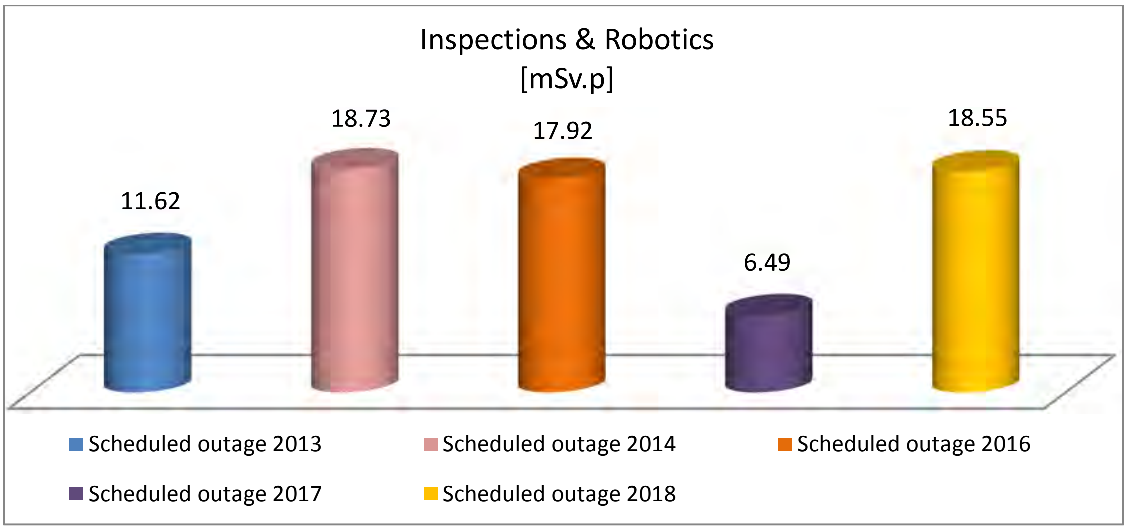
The reactor inspection program requires performing, at each scheduled outage, the metrology of the primary coolant channels (that are transported to the spent fuel storage pool building I) and inspections of the reactor internal components. Dose reduction achieved for the different scheduled outages which is shown below, is due to:

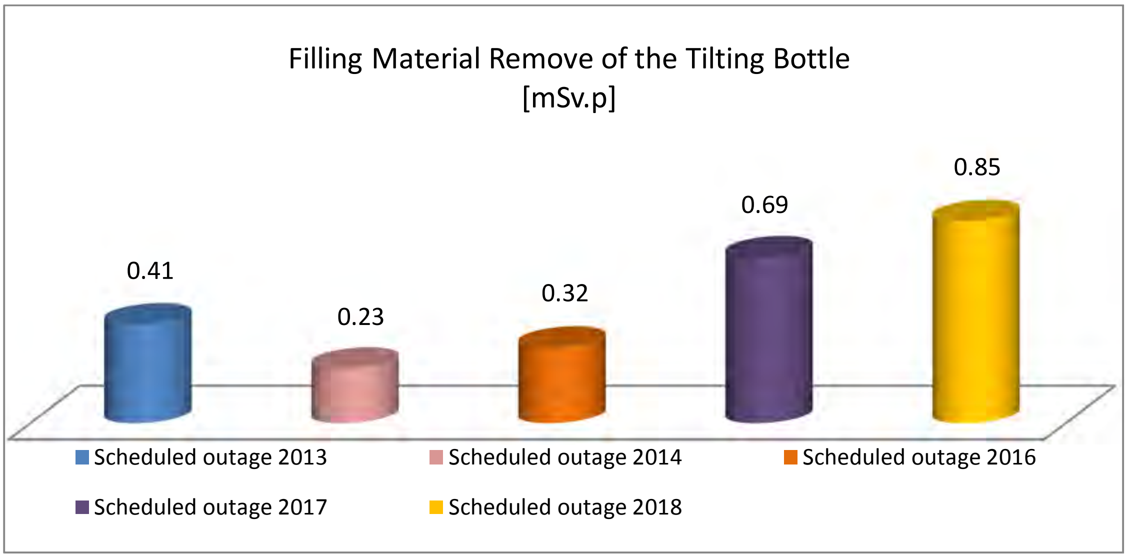
- Shorter inspection times by the use of new tools.
- Increase of control and monitoring of personnel individual doses.
- Optimal inspection staff training.
- High degree of decontamination of tools used for inspection.

- Increased use of pressurized suits to remove reactor inspection cameras.
- Improvements in shielding of work areas near to the intake of the coolant channels.

Below are showed comparison charts for different tasks and sub-tasks to carry out the inspection program.





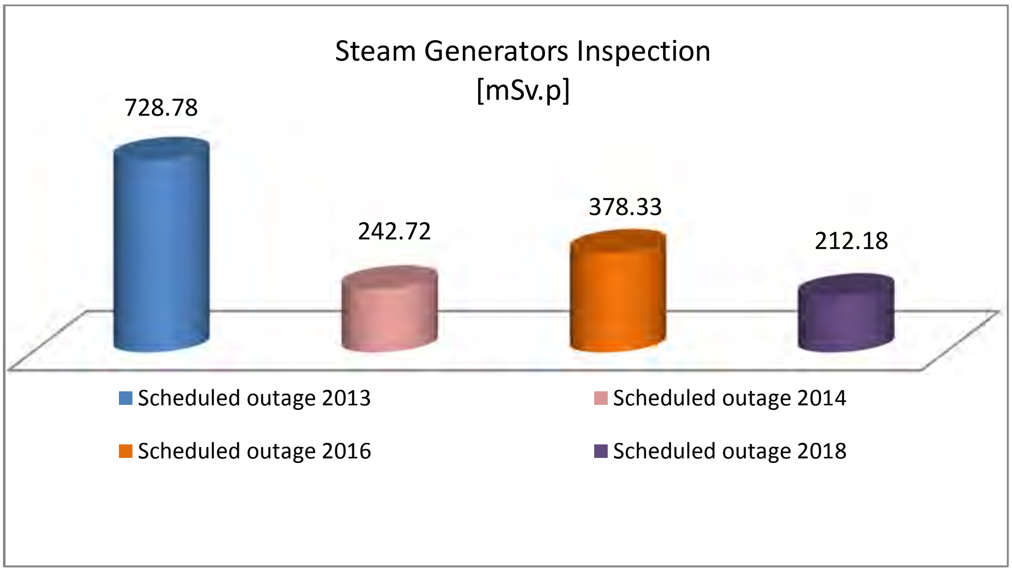


3.15.6.2. CNA I STEAM GENERATORS INSPECTION

Dose reduction achieved for different scheduled outages due to:

- Shorter exposure times in areas with high radiological exposition.
- Better control and monitoring of the personnel individual dose. Use of tele-dosimetry.
- Involved personnel highly qualified.
- Location changing of the racks to less exposed areas (in previous plant scheduled outages they were in front of the boiler plenum. At this scheduled outage they were placed at the main pumps building).

The figure below shows the corresponding comparison chart.



3.15.6.3. CNE COLLECTIVE DOSE MONITORING AND TRENDS TRACKING

During the last three years, the Dose Optimization Division (now known as Optimization of Protection) has extensively worked in reducing doses for CNE staff. This staff, *included not only station personnel divided by areas (Maintenance, Operations, RP & safety Services, Engineering) but also for all other contractors working during the CNE PLEX*

The Collective Doses *estimated at the beginning of the Project, was based on OPEX and work management*. Optimization of Protection division performed tracking/analysis *for* the used dose progression *at CNE and other similar NPPs as well (such as Wolsong 1 in Corea).* By applying the ALARA philosophy and criteria, *it was possible* to achieve the best possible result.

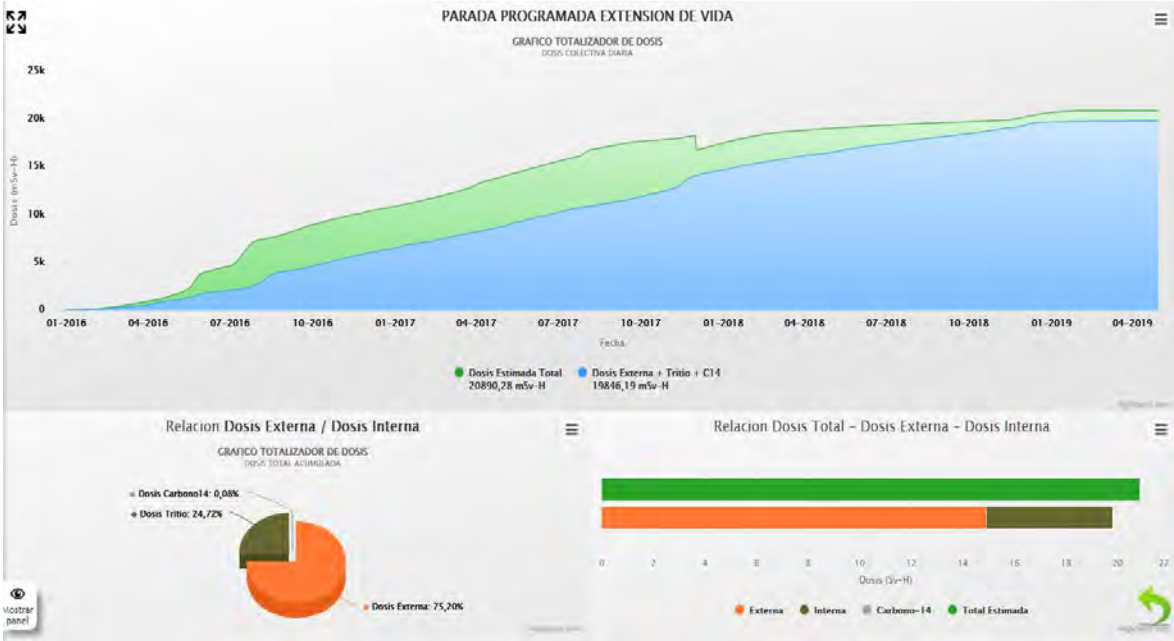
These last three years, *this team* has led the CNE collective dose to have a considerable reduction. Thus, some of the actions being taken were:

- *ON-LINE Dosimetry System Management, avoiding unplanned exposures.*
- Challenging OPEX information for continuous improvement in the dose estimation process.
- *Update* of the Digital RWP.
- Hotspots database Update.
- Tracking and trending analysis for the values obtained from the RWP.
- Shielding planning (permanent and temporary).
- Improved Signage for rooms and systems.
- Staff Training.

All these actions, among others, are part of the 5 years ALARA program. The percentages of improvement *for* the CNE collective dose during *the refurbishment outage were*:

- *2016*: a reduction of *52%* was achieved.

- *2017*: a reduction of *18%* was achieved.
- *2018*: a reduction of *5%* was achieved.



**Estimated (green) vs Actual (Blue) Collective Doses.*

3.15.7. COMPLIANCE WITH THE OBLIGATIONS IMPOSED BY THE CONVENTION

The ALARA activities have been reinforced and intensified with the objective of optimizing the dose received by the personnel occupationally exposed to ionizing radiations, reinforcing its activity with the better implementation of mock-ups, training, improvement in tools and the analysis of the work plans.
Limits for the doses for personnel occupationally exposed have not been exceeded. The annual average value of doses for the public (*representative person*) resulted lower than *0.8%* of the *individual dose limit (1 mSv).*

Therefore the country complies with the obligations imposed in Article 15 of the Convention on Nuclear Safety.

3.16. ARTICLE 16: EMERGENCY PREPAREDNESS

i. *Each Contracting Party shall take the appropriate steps to ensure that there are onsite and off-site emergency plans that are routinely tested for nuclear installations and cover the activities to be carried out in the event of an emergency.*

For any new nuclear installation, such plans shall be prepared and tested before it commences operation, above a low power level agreed by the Nuclear Regulatory Authority.

ii. *Each Contracting Party shall take the appropriate steps to ensure that, insofar as they are likely to be affected by a radiological emergency, its own population and the competent authorities of the States in the vicinity of the nuclear installation are provided with appropriate information for emergency planning and response.*

iii. *Contracting Parties which do not have a nuclear installation on their territory, insofar as they are likely to be affected in the event of a radiological emergency at a nuclear installation in the vicinity, shall take the appropriate steps for the preparation and testing of emergency plans for their territory that cover the activities to be carried out in the event of such an emergency.*

3.16.1. INTRODUCTION

Onsite and off-site emergency plan to respond in case of nuclear or radiological emergency at Nuclear Power Plants (NPPs), is required to the operator by the Regulatory Body. This plan *must contemplate the response actions to be taken within the installation as well as the necessary to be implemented off-site. This plan is a requisite to obtain the operating license.*

The Emergency Plan includes all aspects related to the necessary strategy to control, mitigate and limit the consequences in the event of an emergency and also establishes the automatic measures for the protection of the population and the actions to be implemented by the response organizations.

The main important issues are:

- ARN advises the Executive Branch *and national, provincial and municipal organizations* on the issues of their incidence, including radiological and nuclear emergencies.
- ARN must provide protection against harmful effects of ionizing radiation, even under emergency situations.
- *ARN approves the procedures and emergency plans, including emergency plans developed by local, provincial and national authorities, as well as training plans and* training plans for members of the public near the NPPs.
- *ARN coordinates the representatives from response organizations in relation to the protection actions necessary* in case of a nuclear accident.

3.16.2. ARN RESPONSIBILITIES RELATED TO NUCLEAR EMERGENCIES

The Nuclear Emergency Response System (SIEN) was created by ARN to comply with the provisions of the Act No. 24,804 and Decree No. 1,390. SIEN is the system used by the ARN to respond in cases of nuclear emergencies and coordinate the national, regional and local response organizations to effectively manage nuclear emergencies in preparedness, intervention and recovery stages, in case of a nuclear accident.

In case of an emergency declaration in any of the Argentinean NPP, ARN has the main role of coordinating the incident command, for which the Municipal Emergency Operative Center (COEM) is settled for. The COEM, placed at the surroundings of NPP, is where the automatic countermeasures are implemented, nuclear and radiological assessment is performed, intervention group´s radiological protection and environmental surveillance are managed, among others. The COEM is integrated by representatives of all the intervenient response organizations (as established on emergency plans) which belong to civil organizations (Fire Brigade, Civil Defense, etc.), law enforcement (Regional Police, National Gendarmerie and Coast Guard) and military institutions (Argentinean Army, Argentinean Navy and Argentinean Air Force). These organizations carry out the necessary actions in order to facilitate the application of *the automatic protection measures in the community.*

The Operative Nuclear Emergency Chief (JOEN) is the COEM coordinator. The NPP manager is the first in carrying out the role of JOEN until the ARN officer in charge arrives to the place and replaces him. *The JOEN's role is the coordination of the implementation of the automatic measures and recommends the application of other measures under the concepts of radiological protection.*

ARN´s strategy for responding to a nuclear emergency includes establishing experts groups and a decision making group. The last one operates at the Emergency Control Center (CCE), placed at the ARN´s headquarters located in Buenos Aires Autonomous City. *From there, assistance is given to the COEM and the strategy response is addressed, including radiological and nuclear assessment, radiological protection of intervention teams and environmental monitoring, among other actions.*

ARN is the National Competent Authority according to the "Convention on Early Notification of Nuclear Accident" and in the "Convention on Assistance in the case of a Nuclear Accident or Radiological Emergencies", and the CCE is the National Warning Point according to the IAEA -Emergency Notification Assistance Technical Operations Manual (EPR-IEComm 2012).

3.16.3. CLASSIFICATION OF EMERGENCY SITUATIONS

Emergencies are classified as following:

- General Emergency: Events which lead into a significate risk and require the implementation of off-site urgent protective actions.
- On site emergency: Events which lead into an important decrease of NPP defense in depth level.
- Alert: Events which lead into an unknown level or a decrease of NPP defense in depth level.

When an abnormal situation has occurred in a NPP, the primary responsible must classify the emergency, apply the plant emergency procedures, and in case of General Emergency, notify to response organizations and ARN. In the last one, the emergency procedures define two states:

- Green Alert: is declared when an unusual situation that may affect the NPP safety is detected. In such case, internal emergency procedures are applied and an internal emergency organization is established. *The conformation of* the external emergency response organization (COEM) *is requested and the preparation of the automatic protection measures of the citizens is performed.*
- Red Alarm: Is declared when the release of significative radioactive material quantities to the environmental is imminent. Internal emergency procedures continue and situation involves a prompt notification to COEM, which communicates the alarm to population and implements the automatic protective measures *started in Green Alert.*

During the on-site and off-site emergency, COEM and the primary responsible maintain a constant information exchange.

3.16.4. EMERGENCY ZONES AND URGENT PROTECTIVE ACTIONS

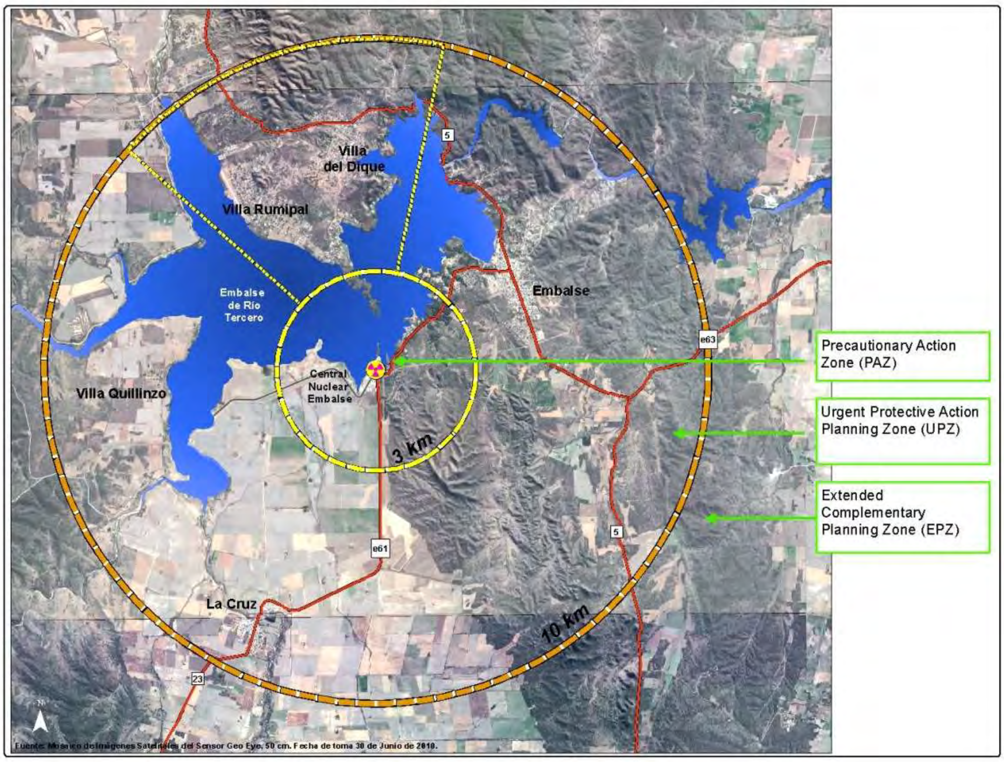
Urgent protective actions and other actions that must be performed once declared the emergency in the NPP are stablished in the approved emergency plans. For that purpose, zones are predefined as following:

- Precautionary Action Zone (PAZ): Is the area enclosed by the 3 km radius from the NPP.
- Urgent Protective Action Planning Zone (UPZ): Defined as the area enclosed between the 3 km to 10 km from NPP. It is being considerate, under IAEA post-Fukushima recommendations, to extent this zone and include the 360°around the NP P.

In addition to these zones, JOEN considers the following zone for the application of other measures:

- <u>Extended Complementary Planning Zone (EPZ):</u> This zone *covers areas beyond the UPZ zone and* is limited by the results of radiological monitoring. *In the EPZ other measures are defined, other than those applied in PAZ and UPZ. Among them, the instruction to reduce accidental ingestion, restriction of consumption of certain foods, decontamination, etc.*

Figure 3.16.1. - Emergency zones map



The urgent protective actions include:

- Early evacuation: Is established at PAZ once declared the *Green Alert* at NPP. *This action is carried out by a Security Force assigned in conjunction with the Civil Defense.*
- Access control: Access control points are established at strategical ingress/egress spots for vehicles, in routes beyond 10 km from the NPP. In a Green Alert situation regional police is prepared to start the access cut. At Red Alarm it is implemented allowing the access to this zone only to response groups.
- Thyroid prophylaxis with stable Iodine: It is implemented at UPZ. *A Security Force assigned* is the organization responsible for the iodine pills distribution at ground and islands. The pills are distributed at *NPPs* and on strategical spots inside the involved towns according to the emergency plan. *In a Green Alert situation the populations receive the pills but the ingestion occurs when Red Alarm is declared.*
- Sheltering: It is implemented at UPZ. In Green Alert situation, population prepares for sheltering remaining inside buildings and when Red Alarm is declared, sheltering is fully implemented by closing doors and windows, sealing holes and turning off *the air conditioning devices, extractors, etc. that intake air from the outside.*

The population remains informed about the measures implementation through the local media, public alert system, sirens, and loudspeakers, among others. This is a main topic at population and response organizations trainings.

Environmental monitoring starts once the release of radioactive material has finished *and is important in the definition* of other protective actions that are detailed below:

- Evacuation of affected zones due to ground deposition.
- *People* decontamination.
- Instructions for reducing inadvertent ingestion.
- Recovery of evacuated zones.
- Iodine prophylaxis.
- Restriction of local produced food.
- Food intervention: *The levels of intervention approved by the ARN for the substitution of contaminated food products were obtained through an optimization analysis. Due to the characteristics of the country, contaminated food may, in general, be replaced by other products from areas not affected by the accident.*
- Ground decontamination, infrastructures, buildings, etc. The execution of this action will be decided in a case by case analysis.
- *Food intervention for commodities: There are intervention levels for food products consistent with the Codex Alimentarius (FAO / WHO).*

3.16.5. NPP EMERGENCY PLAN

NPP´s emergency plans include the necessary information for preparedness, *mitigation* and response in case of nuclear emergencies

Among the topics developed are:

- Agreements with intervening organizations in order to implement the protective actions.
- Responsibilities and functions of the organizations during the response.
- The arrangement, responsibilities and specific function of the NPPs Internal Committee of Emergency Control.
- The following actions to face an emergency situation:
  - Emergencies Response System Activation.
  - Situation Assessment.
  - Beginning and finalization of protection actions.
- Communication protocols needed for emergencies management.
- Detailed protection actions, taking into account the accident and its possible evolution, mainly in the following cases.
- The way of implementing protection actions
- Communication protocols for alerts, information or instructions to potentially affected population (broadcasting, TV, loudspeakers, alarms, etc.)
- Protocols of dose control for emergency response group dose.
- *Training and external exercises with the community and response organizations.*

3.16.6. EMERGENCY PLANS OF RESPONSE ORGANIZATIONS

At the present, there are emergency plans already in force that were approved by ARN, mainly in the municipalities involved in the predefined emergency zones. However, ARN is currently working with the National System for Comprehensive Risk Management and Civil Protection (SINAGIR) in the development of a plan that covers all areas for the postulated nuclear accident scenario.

Otherwise, ARN is also working with response organizations in the creation and updating of action protocols for the specific tasks of their concern.

Following the stipulations of the legal framework, the guidelines for the plans, protocols and others are established by ARN. Within the framework of the plans, agreements and other arrangements are established between different organizations. These agreements allow the use of networks, communication systems and measurement and response equipment of these organizations, among others.

Agreements with national organizations such as the Argentine Army, the Argentine Navy, the National Gendarmerie, the Argentine Naval Prefecture, the National Meteorological Service, the Ministry of Health and the Provincial and Federal Police ranging from framework agreements to assistance protocols, are reviewed and exercised in the simulations of the Nuclear Emergency Plan.

3.16.7. EMERGENCY PLAN EXERCISES

Annually, NPPs perform internal emergency exercises focusing on the assessment of response groups and plan improvement. On the other hand, the exercises which involve the public and external organizations participation are twice-yearly and its scope must be agreed with ARN.

The main objectives of emergency exercises are:

- To expand diffusion of updated procedures between all the intervenient organizations.
- To establish command systems and verify the capability of protective measures implementation (sheltering, evacuation, thyroid prophylaxis with stable iodine, access control, notifications, messages to public, etc.)
- *To strengthen the diffusion to members of the public with the planned actions and encourage their active participation in the exercises.*
- *To test the activation protocols of each intervening response organization and verify that it is updated by incorporating the findings observed in the previous exercise.*
- *To strengthen and reinforce* the trainings on emergency for intervenient response organizations and local community.
- To identify needs and weaknesses and incorporate them as opportunity for improvement.

Nuclear emergency exercises performed since last Nuclear Security Report, involving population and response organizations, are the following:

- *Embalse NPP Exercise - Villa del Dique and Villa Rumipal, Córdoba Province, September 14th, 2016.*
- *Atucha NPPs Exercise - Zárate, Buenos Aires Province, November 2nd, 2017.*
- *Embalse NPP Exercise - La Cruz and Villa Quillinzo, Córdoba Province, September 27th, 2018.*

Apart from the operator and ARN, the following organizations have actively participated and received the specific training for each performed exercise:

- *SINAGIR.*
- National Army.
- National Navy.
- National Gendarmerie.
- National Coast Guard.
- National Air Force.
- *Argentine Federal Police.*
- Buenos Aires Police.
- Córdoba Police.
- Córdoba Civil Defense.
- Zárate Civil Defence and Lima Sub branch.
- Embalse Civil Defense.
- *Villa del Dique Civil Defense.*
- *Villa Rumipal Civil Defense.*
- *La Cruz Civil Defense.*
- *Villa Quillinzo Civil Defense.*

- Embalse Fire-fighter Brigade.
- *Villa del Dique Fire-fighter Brigade.*
- *Villa Rumipal Fire-fighter Brigade*.
- Lima Fire-fighter Brigade.
- Zárate Fire-fighter Brigade.
- Local FM stations.
- National Meteorological Service.
- Local Educational Institutions.
- Private institutions.

All the exercises are carried out with the participation of local resident population that lives to 10 km around the NPP and all urgent protective actions are practiced. Information diffusion and population preparedness are key tasks to reduce as minimum as possible the nuclear accident consequences

Therefore, previous to each exercise, diffusion activities and trainings to local educational institutions are carried out. School trainings are divided in three levels, according to student's ages:

- Level 1, for children from 4 to *7* years old.
- Level 2, for children from *8* to 12 years old.
- Level 3, for teenagers from 13 to 18 years old. This talk can be adapted and it´s used for adults schools trainings (students from 18 years old onwards).

At the same way, the protective actions diffusion is performed to other population areas through interviews, debates and community talks, organized by local Civil Defense, the NPP and ARN.

Protective actions, diffused on preparedness stage, are practiced on an effective way during the exercise. On this way, all level students perform sheltering on local schools and take the iodine pills (replaced by candies). *The Security Force is assigned for the distribution of* iodine pills to population at UPZ. It is also practiced at the exercise. Likewise, all involved response organizations test the efficiency of their functions fulfillment during this event

During this report period, at the Emergency Implementation Exercise, *response scenarios were incorporated, particularly, monitoring scenarios in emergencies with the objective of training and interaction of response teams.*

Year after year, a greater participation of the population close to the NPP is observed, giving more importance to the exercises.

In all emergency exercises, ARN achieved the objective of leading all the organizations response groups.

In addition, each one of the exercises is a learning process to improve the necessary skills and strengthen to work together with the organizations involved in the emergency.

3.16.8. BILATERAL AGREEMENTS AND EMERGENCY CONVENTIONS

In 1986, Brazil and Argentina signed the Argentine-Brazilian Co-operation Agreement. This agreement includes the Chapter “Co-operation and Mutual Assistance in Cases of Nuclear Accidents and Radiological Emergencies”. Brazil is the only neighbor country with NPPs.

Argentina, Chile, Bolivia, Uruguay and Brazil are Contracting Parties of the Convention on Early Notification of a Nuclear Accident, framework to facilitate the exchange of information in the event of a radiological emergency due to a nuclear accident. In the framework of these agreements, representatives of Brazil and Uruguay took part as observers in some of the exercises carried out in Argentina.

3.16.9. CAPACITIES IN THE PREPARATION FOR EMERGENCIES IN ARN

For the intervention in radiological emergencies in facilities, public roads or during the transport of radioactive material, the ARN account with an Intervention System in Radiological Emergencies (SIER) which complements the SIEN. The functions of both SIER/SIEN systems contemplate:

- *Advise those responsible for the facilities in case of radiological / nuclear emergencies.*
- *Advice public authorities involved in the control of radiological / nuclear emergencies.*
- *Intervene in emergency situations in those facilities or practices where radiological accidents occur that cannot be controlled by those responsible for them, or involving members of the public.*

Following a specific procedure SIER / SIEN Guard System is activated according to the characteristics of the emergency. The staff of SIER / SIEN Guard includes 24-hour on-call personnel throughout the year, who are trained and experienced in various fields of action. The System can be immediately activated and operate in such a way that the response is effective and in accordance with the nature and magnitude of the emergency.

In the last years, staff members have joined the guard and have been trained in emergency management. Training was strengthened in relation to operative capacity of the radiological emergency response system by using the opportunity of different events planned in the country, such as events with a mass audience.

SIER / SIEN have various devices for emergency response, including radiation and contamination measurement equipment, personal protection elements, logistics and support equipment.

On the other hand, the *Emergency Control Center of the ARN has a series of computational tools for the modeling of scenarios in case of a radiological or nuclear emergency with dispersion of radioactive material. In addition, ARN has a Geographic Information System (GIS) as a versatile tool for planning the response in cases of nuclear emergency.*

GIS can integrate geographical information in a database, such as social, economic, physical and environmental details, with atmospheric dispersion and ground deposition models which allow prediction of environmental impact during the emergency situation. This enables real time data assessment which certainly facilitates the decision making process.

GIS contributes to establish potential scenarios in emergency situations, but also to identify which one is the most probable and its possible effects. Therefore, the most suitable actions for each zone can be determined. These actions include alternative evacuation routes, appropriate reception center for evacuated population, monitoring team deployment, aerial measuring system operating areas, people and vehicle decontamination spots, etc.

During 2015 ArcGIS online was added to the existing tools. As it allows accessing GIS information from any device connected to the Internet, it will be an important mean for evaluation groups at the CCE and for the JOEN along with his advisors at the COEM, but also for monitoring teams at the field.

ARN holds three different consequence evaluation models for nuclear emergencies. The self-developed Accidental Dose Assessment System (SEDA), US-NARAC's International Exchange Program (IXP) developed for DOE's National Nuclear Security Administration (NNSA), and finally WMO's Regional Specialized Meteorological Centers' (RSMC) models. Evaluation models results are integrated to GIS to extend the analysis with available database.

SEDA is a Gaussian atmospheric dispersion model based code which can be applied to radioactive releases from NPPs. Data entries include meteorological data, such as wind speed and direction, cloud coverage, temperature vertical slope and atmospheric stability class, and also information of released isotopes to the atmosphere (cuali-cuantitative estimates). The kind of data resulting from this code is isodose and isoconcentration lines which help to define protective actions.

IXP is a real time prediction system for radioactivity transportation in the atmosphere in case of nuclear accidents. It is based on NARAC's web version software for time-dependent calculations on atmospheric dispersion and dose assessments. It has a meteorological prediction model associated, but also the possibility to enter real climatological data. As a result, different models can be used to estimate the impact of a release in an emergency situation.

Through Argentinean Meteorological Service (SMN) assistance from World Meteorological Organization is obtained. WMO then transmits the request to Regional Specialized Meteorological Centers (RSMC) to run their models to estimate dispersion at a global scale.

As it was previously noted, a favoring circumstance is that results obtained from the different models can be integrated to the geographical information system. This allows cross-referencing of potential scenarios to execute actions previous to, during and after an emergency situation.

Emergency Control Center, located at ARN's headquarters, has recently enhanced its communicating system. It includes an independent Local Area Network, which is capable of broadcasting videoconference sessions through the Internet, and an Integrated Digital Service Network, but also for

GIS server capable of transmitting processed information at the CCE to the COEM. Moreover, a satellite mobile system allows sending information from anywhere on the field to the CCE.

With the objective of enhancing communication systems, ARN worked together with NPPs emergency departments in merging communications via satellite. This way, ARN and NPPs have the same technology which is highly advantageous.

3.16.10. IMPROVEMENTS CONCERNING EMERGENCY PREPAREDNESS IN THE NPPs

As a result of the analysis and assessments carried out as part of the stress tests carried out in 2011/2012, analyses of emergency drills and exercises results, and areas for improvement identified during regulatory audits, self-assessments and external audits, the following improvements were implemented.

3.16.10.1. CNA I AND CNA II

- An environmental monitor net around the plant which consists of mobile and fixes radiological and meteorological stations was implemented to facilitate an efficient response in accidental situations by having real time information.
- Source term:
    - A verification of the reactor and the spent fuel pools source term was performed.
    - Update the source term of the calculated scenarios with radiological consequences for all conceivable types of severe accidents at the site. This improvement will be carried out when the PSA level 2 is available.
- Six new fix and mobile satellite phones, available to be used as a back-up of the existing communication systems, were provided.
- Procurement of a foam and air system against fire (cannon for long distance action) installed in a mobile foundation in the heliport zone which can eventually contribute to the washing and the retention of radioactive particles was installed.
- A new Internal Centre for Emergency Control was built. The Centre was designed and built taking into account the recommendations and suggestions provided by WANO in the 2012 Peer Review and in the Emergency Preparedness Technical Support Mission that took place in 2013, as well as the proposals that came from performed benchmarking.
- *There is a dedicated and exclusive organization to deal with emergencies. This emergency response organization (ORE) is made up of 94 positions and each position is made up of a minimum of 3 to 4 people. All the staff that integrates this organization is owned by the site.*
- *There are Emergency Procedures and Severe Accident Guides to deal with various situations that may occur on the site.*
- *There is Procedure PS-E-01 "Conformation and operation of the Emergency Response Organization (ORE)" where the organization to deal with an emergency involving the Atucha site and the public, up to a radius of 10 km is defined. This procedure defines the roles of emergency organization posts, alert and emergency organization service, as well as interaction with external response organizations, for example, Municipal Civil Defense, Nuclear Intervention System Emergency and the Regulatory Body (ARN). In the case of an emergency, it is a maximum period of 60 minutes to arrive to the site.*
- *A new Municipal Emergency Control Centre has been planned far from the nuclear emergency planning area (more than 10 km) taking into account Fukushima's recommendations regarding infrastructure, instrumentation and communication systems. The place where the centre will be located will be defined and detailed engineering is in progress.*
- *Emergency exercises are carried out, where the time has been extended to improve processes and communications, being more realistic and complex, achieving greater challenges for the personnel of ORE.*
- *Four emergency exercises are conducted per year. Each person involved in the Emergency Response Organization (ORE) must participate in at least one exercise per year. Of those four*

exercises, one involves all the staff of the site. In addition, during the whole year there are simulations of partial communication and simulations of emergency brigades of units I and II.

- *In 2018, exercises to summon the available guards of the ORE were incorporated into the annual exercise planning, where the arrival times of the CNA are measured.*
- *Based on the observations made by WANO, in the year 2017 changes of guard were implemented and trained in one of the exercises carried out during the year.*
- *The exercises are designed for both units. Exercises are carried out where one of the affected units is simulated and the other in operation or both affected plants. Every year, different scenarios are carried out to test team capacities and staff skills.*
- *Every year emergency exercises are carried out in the simulator for all Operations Guards of both Units. These exercises were carried out in the simulator for the operators of the control room and in the field for the field operators, including fire exercises.*
- *The procedure PS-E-21 "Emergency Exercises" is available, it follows the guidelines of GSR part 7 in order to establish the process to be followed during the exercises and emergency drills.*

3.16.10.2. CNE

- *CNE has a network of 12 radiological monitoring stations to be used in Normal Operation and in emergency in the plant, which facilitate an efficient response in accidental situations by having information in real time.*
- *Source term:*
  - *A verification of the reactor and the spent fuel pools source term was ended.*
  - *The source term of the calculated scenarios with radiological consequences for all conceivable types of severe accidents at the site was updated. This improvement was carried out with the PSA level 2 which it is available.*
- *CNE is provided with 5 fix satellite phones.*
- *The re-evaluation of all applicable procedures in the emergency management strategy was carried out and new procedures were generated.*
- *A review of the emergency plan was made for the new cycle of operation of the Plant.*
- *To meet the needs of different jobs outside the buildings during an emergency, the following equipment was acquired:*
  - *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.*
- *The review and improvement of the procedures applied to the co-ordination and reception of supplies, equipment and additional personnel involved in the emergency management was implemented.*
- *Emergency exercises expanded over time were implemented. The exercises aim to have long-term operational emergency centres to improve the emergency management transfer mechanism, the sustainability of resources, the provision of technical support, communications and the acquisition of trustworthy data.*
- *CNE incorporated 2 tanks of 8,000 litter (with motor pump, motor generator and submersible pump), 1 tank of 4,500 litters (with mechanical pump) and 1 tank of 3,000 litters to provide water used in different tasks during the emergency.*
- *A conditioning of a municipal emergency control centre will be carried out far from the zone of nuclear emergency planning (more than 15 km). The centre will have infrastructure, instrumentation and communication systems according to what was learned in the Fukushima accident.*
- *The building of the Internal Emergency Control Centre (CICE) was modified and improved. A ventilation system with HEPA / active carbon filters and emergency electric power was installed. The communication resources were expanded in it and a new communication room was created.*
- *The building of the External Emergency Control Centre (CECE) was modified to provide a more effective response to emergencies.*

3.16.11. COMPLIANCE WITH THE OBLIGATIONS IMPOSED BY THE CONVENTION

Formal and operative activities arisen from the Early Notification and Assistance Convention kept going normally. Argentina participates in emergency events and in CONVEX international exercises.

ARN is the designated national competent authority following the Articles of both conventions. Its Emergency Control Centre (CCE) is the National Contact Point for Alerts in accordance to the Operations Manual for Incidents and Emergencies Communications of the IAEA (EPR-IEComm 2012), mainly through the USIE platform.

ARN participates in the meetings of the Representatives of Competent Authorities identified under both conventions every two years. In this context, ARN has presented its experience in different cases in relation to intervention in emergency events.

Regarding the implementation of the Article 2, Paragraph 4 of the Assistance Convention in relation to the agreement among Contracting Parties to cooperate between themselves and with the IAEA to facilitate the timely provision of assistance in the case of a nuclear accident or radiological emergency, in order to mitigate its consequences, and to the response and assistance network (RANET) developed among the signatory states of the convention, Argentina participates in this network through the ARN Biodosimetry laboratory, since 2008, with its capacities on Dose Assessment (DA-1 Cytogenetics-based biodosimetry, Fat and EBS).

On the other hand, Argentina participates in the international CONVEX exercises coordinated by the IAEA, in the framework of both conventions. In these exercises, ARN tests its EPR capacities, registered or not in the RANET.

Preparation activities for emergencies related to nuclear safety and security in NPPs developed as planned. Focus was put on disseminating radiological risks and emergency plans to the population in the surroundings of NPPs.

Hence, it can be concluded that Argentina have up-to-date emergency plans capable of facing different situations at nuclear facilities, in which actions not only for the onsite, but also for the offsite, are described. Periodic exercises are carried to test implementation of these emergency plans.

Argentina fulfills all the obligations impelled on article 16 of the Convention of Nuclear Safety.

3.17. ARTICLE 17: SITING

Each Contracting Party shall take the appropriate steps to ensure that appropriate procedures are established and implemented:

i. for evaluating all relevant site-related factors likely to affect the safety of a nuclear installation for its projected lifetime;

ii. for evaluating the likely safety impact of a proposed nuclear installation on individuals, society and the environment;

iii. for re-evaluating as necessary all relevant factors referred to in sub-paragraphs (i) and (ii) so as to ensure the continued safety acceptability of the nuclear installation;

iv. for consulting Contracting Parties in the vicinity of a proposed nuclear installation, insofar as they are likely to be affected by that installation and, upon request providing the necessary information to such Contracting Parties, in order to enable them to evaluate and make their own assessment of the likely safety impact on their own territory of the nuclear installation.

3.17.1. INTRODUCTION

The objective of the siting studies is to select a suitable site for a NPP, including appropriate assessment and definition of the related design bases, taking into account that NPP's design implies the consideration of site dependant factors which may affect, directly or indirectly, the plant safety. For instance, the capability and reliability of the ultimate heat sink and power supply networks, the potential occurrence of natural and/or man induced events, and the characteristics of communication routes and accesses.

Therefore, the siting studies aim at determining the effects of external events occurring in the region of the site, to evaluate the potential radiological impact on the environment due to the plant operation and the feasibility of the emergency plans.

In Argentina, the studies (for selecting the location of a NPP) are part of the requirements the Licensee shall comply with at the time they request a construction license, or included in the Periodic Safety Review (PSR), document necessary to require the Operating License renewal. A previous and independent licensing of a site is not explicitly required.

The results of siting studies of the NPPs were used in determining parameters required for the application of models describing radionuclide dispersion to the environment. These models enabled the evaluation of dose exposure due to radioactive effluents released during normal operation.

Moreover, the information supplied by siting studies enabled to foresee the implementation of actions required to protect the public from accidental situations. These actions were taken into account in the elaboration of the corresponding Emergency Plans.

3.17.2. EVALUATION OF SITE RELATED FACTORS

3.17.2.1. NORMATIVE ASPECTS

In Argentina, a NPP construction must not be initiated without a previous Construction License issued by the Regulatory Body, upon request from the Licensee.

In line with this approach, at the time of applying for the Construction License, the Licensee must submit to the Regulatory Body all the documentation required to evaluate the radiological and nuclear safety of the installation to be built, including the site characteristics in relation to:

- Natural and man-induced external events that could affect the installation safety.
- Dispersion of radionuclides to the environment, both in normal and accidental conditions.

The Regulatory Body issues the license once the Licensee has demonstrated that the design of the NPP to be built complies with the regulatory standards and other specific regulatory requirements for the selected site, taking into account the NPP-site interaction.

Besides, the Regulatory Standard AR 10.10.1. “Site Evaluation for Nuclear Power Plants” had recently been developed and put into force, taking into account the lessons learned from the Fukushima accident and the corresponding IAEA standards.

3.17.2.2. EXISTING SITES

Two sites were at the time selected and evaluated as suitable for NPPs construction in Argentina:

- Atucha, on the right bank of the Paraná de Las Palmas River, in the Province of Buenos Aires, and;
- Embalse, on the coast of the Tercero River Dam Lake, in the Province of Córdoba.

At Atucha site there are two independent units, CNA I and CNA II in operation, and CAREM 25 is under construction. *At Embalse site, the CNE refurbishment process finished in December 2018 and the start-up began in January 2019, reaching 100% of its full power by end of April 2019.*

The site studies performed for CNE and CNA underwent in the following three stages:

- Survey of the region of interest,
- Selection of the candidate site, and, finally,
- Evaluation of the selected site.

For each NPP, the first stage was the survey of an extensive area with the purpose of screening (accepting or rejecting) those locations that could be candidates for location of a NPP. At this stage not only were safety considerations taken into account but also their economic and social aspects, as well as their evolution perspective during the plants lifetime.

3.17.2.3. SITING STUDIES

3.17.2.3.1. Siting studies performed

The original siting studies related to CNA I and CNE locations were fully detailed in the previous Nuclear Safety Convention reports. The most significant external hazards affecting the design basis were seismic events for the CNE site (earthquakes and geological faulting) and hydrological events for the CNA site (extreme values of the Paraná River flooding and low level water). Besides, tornado hazards were considered for both sites Atucha and Embalse.

With regards to CNA II, due to the fact that the unit is located in the same site as CNA I, specific information was available at the time of the CNA II design stage. This was the result of continuous studies that are being carried out for CNA I since it first began operating, particularly about hydrological, tornadoes, extreme meteorological phenomena and atmospheric dispersion, and population distribution aspects, as well as to the nuclide transfer mechanisms models, both through atmospheric and aquatic media.

Additionally, specific site related studies were carried out to determine the design basis parameters for CNA II.

Extensive geotechnical investigations were performed to define the soil characteristics for stability verification and foundation design, as well as geophysical investigations and studies to determine the seismic hazard in accordance with new criteria and data. Thus, the report entitled "Seismic Study of CNA II NPP Siting" reflects the results of these investigations, which were carried out in 1981 by the "Argentine Nuclear Company of Electrical Power Plants". It should also be mentioned that the corresponding chapter of the "Preliminary Safety Analysis Report" was issued in 1981 and included all the information about the site.

Nevertheless, the site of CNA II is of very low seismic risk, seismic design principles and construction measures were applied to CNA II NPP to withstand horizontal loads typical of regions with low seismicity. According to seismic design criteria for regions of low seismicity these values used for design are not related to the characteristic site design basis earthquake.

Another natural external hazard that has been included in the design basis of the plant is a tornado.

Accordingly, the impact of matter projected and the pressure loads of a tornado were also considered in the design of the plant. It is worthwhile mentioning the operation of a meteorological station (with a 100 m tower) at the site, which provides specific site micro-meteorological data.

Man induced external events are also considered in the plant design basis. In this regard, an explosion pressure wave corresponding to deflagration of a gas cloud is defined, which is duly taken into account through appropriate layout and structural aspects. Also, measures to avoid the entering of explosive gases into buildings and structures are taken into account in the design of the plant.

Other site related studies to be mentioned are the specific population census in the plant surroundings (completed by regional and national census) or those related to the cattle breeding and agricultural production in the vicinity of the plant.

Moreover, values of tritium concentration measurements in environmental compartments were confronted against the corresponding theoretical values, enabling the dilution factors validation.

Finally, it should be mentioned that monitoring on vegetables, atmospheric and hydrological sediments, cow milk from the region, fish, water, etc., are carried out, which enable the verification of transfer parameters that are used in the evaluation of individual and collective dose associated with CNA II operation.

As was mentioned, the CAREM-25 prototype reactor is under construction in the Atucha site, next to CNA I and CNA II. The main requirements for the site selection are related to the protection of the public and the environment from the radiological consequences due to accidents and their mitigation in case they should occur.

The site selected includes the necessary infrastructure and effective security measures with an established response force which has proven its competence to handle emergencies effectively during annual drills.

The site is suitable for building the CAREM as was demonstrated during the many studies carried out during the CNA I Project. These studies undertaken by NA-SA (Nucleoeléctrica Argentina S.A.), duly extrapolated and updated were made available to CNEA (National Atomic Energy Commission of Republic Argentina), responsible for the design, construction and operation of the CAREM prototype reactor. These studies include, inter alia, external hazards, population density & distribution, and NPP lay out.

These studies were complemented by others related to the CAREM Project and its specific location, such as the geological studies and the impact of CAREM on CNA I and CNA II, and vice versa.

This site selection facilitates communication with the community and a co-operation program to assist its needs, contributing to its social and economic development. At least at its initial operating stage the CAREM prototype will not supply electrical energy to the national grid.

Agreements have been signed to ensure co-operation and feedback of experience between CNEA and NA-SA in connection with radiological and nuclear safety, physical protection, security, exchange of technical information including environmental monitoring data and the balance of plant design.

3.17.2.3.2. Site re-evaluation

The selection of CNA I, CNA II and CNE sites, were performed, according to the criteria and the information available, during the sixties and the early seventies.

Nowadays, there are more accurate tools to define the design parameters than those existing at that time. Therefore, among others, the following reasons make necessary to re-evaluate these parameters including an analysis of its impact on the current safety condition:

- Existence of new and more reliable data and information for assessing the occurrence of external event hazards, in comparison with those available at the time of the original design.
- Availability of Methods and criteria consolidated through their use in different regions (i.e. different natural ‘environments’) and countries (i.e. different regulatory and practice ‘environments’) allowing to reach certain consensus in the international practice about what to do and how to do it.
- A more balanced situation between the regulatory requirements and the industry practices, through the use of more realistic and integrated criteria, with a trend to reduce the use of excessive conservatism.
- Changes in both the real plant configuration and the present conditions in the nearby region; as well as those modifications introduced to structures, systems and equipment to improve their performance.

The design parameters corresponding to earthquakes, extreme meteorological phenomena and man induced events were determined according to the region and site specific conditions of each plant.

The Argentine's NPPs seismic design is consistent with the national regulation, AR 3.10.1. "Earthquake Protection in Nuclear Power Plants", and international criteria established at the NPPs design time. However, since these regulations have been updated over time, the NPPs have faced a seismic safety assessment (SSA) by using the seismic margin assessment methodology (SMA) against the occurrence of a certain level earthquake (Review Level Earthquake, RLE) higher than the design basis earthquake (DBE).

To perform the above mentioned SSA, the Licensee of CNA I; CNA II and CNE updated the seismic hazard of each site using methodologies and databases according to the state of the art (based on the approach of the US NRC Regulatory Guide 1.208).

As a consequence of the Fukushima accident and in order to apply the corresponding lessons learned, the Regulatory Body requested perform a stress test to each Argentinian NPP consisting in a reassessment of the NPPs safety margins assuming the occurrence of a sequential loss of the lines of defence in depth caused by extreme initiating events and, among others safety related issues, it includes:

a) Extreme initiating events conceivable at each NPP site.

b) The loss of safety functions caused for each one of the extreme initiating events considered.

c) Arrangement / disposal of SSCs belonging to safety systems to assure they can continue fulfilling the corresponding safety function.

d) 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 are reached. This is to identify the most adequate recovery strategies and the components that must be available for each of the corresponding strategy implementation.

e) Safety implications derived from multiple reactors located in the same site, identifying and implementing the corresponding measures and the procedures to use the existing resources of one unit to assist another unit.

In response to the mentioned regulatory requirement, the CNA I, CNA II and CNE Licensee performed the required stress test and submitted to the Regulatory Body the corresponding Stress Test Reports. Later on, the Regulatory Body carried out an assessment of these reports to verify compliance with the provisions of the regulatory requirement. As a result, many opportunities for improvements have been identified, but in no case were there weaknesses or critical situations (cliff edge effects) found, which would make necessary to take urgent actions.

The improvements and modifications proposed by the NPPs Licensees included an implementation schedule composed by short, medium and long term actions.

As a result of the stress tests performed by the Licensee and the corresponding assessment carried out by the Regulatory Body, it was issued a regulatory requirement asking additional assessments or the implementation of improvements and modifications referred to seismicity, flooding and other extreme external hazards, many of which have already been implemented.

Further on there's a summary of the re-evaluation of siting aspects at different times after the original evaluation, as well as those being carried out at present.

3.17.2.3.2.1. CNA I and CNA II site re-evaluation

For Atucha siting, the following re-evaluations were performed:

- Re-evaluation of tornadoes and severe storms hazard, including the energy transmission lines which are essential for the plant safety. A working plan was prepared, starting in December 1998. A complete database has been compiled adding the 20 years of additional data, since 1980, and a more refined model for assessing the tornado impact probability was used (i.e. Twisdale and Dunn).

- Evaluation of the resistance of important buildings for nuclear and radiation safety against missile impact generated by tornadoes for CNA I (2014).

- Concerning the re-evaluation of potential hazards from man-induced events at the plant site, the document of Design Basis Threat *has to be periodically* reviewed by NA-SA, and experts from the national organization in charge of the physical protection of the NPPs undergo continuous training.

- Different alternatives have been studied regarding flooding. Actually not only the water-level but also other events resulting in intake channel blockage have been studied. Operating instructions have been developed by listing the actions that have to be taken depending on the water-level measured in the intake channel. New hydrologic and hydraulic studies of CNA I / CNA II *were* carried out.

- Those instructions take into account the configurations needed to keep the plant in a safe situation, considering also the cooling of the spent fuel pool. The second heat sink is an improvement regarding safety for these scenarios, providing an independent system from the residual heat removal system, which uses the river as the final heat sink.

The activities related to assessments or improvements arisen from the stress tests results include the following:

3.17.2.3.2.1.1. Earthquakes

Atucha site is located in a region of low seismicity. The SSA based on the corresponding updated seismic hazard is being carried out since 2011 and the objective is to demonstrate the seismic safety margin for CNA I and CNA II for the RLE and to confirm the non-existence of cliff edge effects.

The updated seismic hazard is based on the comparison of two Probabilistic Safety Hazard Analysis (PSHA) performed by AECL and JJJ & Associates on behalf of the Licensee, which follows the principles outlined in the IAEA Safety Guide SSG-9 "Seismic Hazards in Site Evaluation for Nuclear Installations".

The maximum Peak Ground Acceleration (PGA) for Atucha site was estimated in 0,1 g, being adopted as RLE having a sufficient margin over the original DBE in order to improve the safety and find weaknesses that could limit their capacity to support the consequences caused by an earthquake greater than the DBE. Based on the PSHA outcomes, the RLE adopted for SSA of CNA I and CNA II is derived from the mean uniform hazard response spectra (UHRS) for a recurrence period of 10,000 years.

The CNA I NPP was not originally designed or qualified considering severe earthquakes. However, due to the conservative design applied as well as the SSCs robustness, it was considered that there is an inherent capability to withstand earthquakes of a certain level which will be determined by means of a SMA to assess the SSCs’ status in relation to their ability to perform its safety function after a specific earthquake occurrence.

The CNA II original design criteria were based on a 0.05 g PGA DBE. Additionally, design principles and construction measures for low seismicity regions were applied.

Considering the above, the Licensee decided to conduct a SMA evaluation to determine the NPPs’ capacity to deal with beyond DBE, based on the EPRI NP6041-SL, “A Methodology for Assessment of Nuclear Power Plant Seismic Margin ".

The SMA methodology aims to determine seismic capacity of high confidence for the NPPs 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.

This methodology considers that HCLPF capacity is the seismic level (SL) for which there is a 95% confidence that the probability of safety systems availability be more than 95%. The results are conservative because the loads used for the SSCs verification are those generated by the RLE simultaneously combined with normal operating loads.

CNA I

The CNA I specific SSA consists of five phases, namely:

- Phase 1: Scope of the study and preliminary plant inspection walkdown.
- Phase 2: Development of the Safe Shutdown Equipment List (SSEL) and Systems walkdown.
- Phase 3: Seismic response and in structure response spectra determination.
- Phase 4: Inspection walkdown of the seismic capacity and screening process.
- Phase 5: Structure capacity evaluation and detailed analysis.

Phase 1 was completed in December 2011. Phases 2 to 4 were also completed in May 2013. Phase 5 was completed during 2015. It comprised a complementary seismic walkdown, screening process results and outliers resolution.

In April 2015 a complementary walkdown was performed; covering a total of 148 items corresponding mainly to the new emergency electric system.

The seismic capacity (HCLPF) was calculated for some outliers, i.e., items that required calculations to determine their HCLPF values compared to the Review Level Earthquake (RLE).

The results of the walkdown were presented in 6 categories:

- Category 1 corresponds to those items for which easy fixes can be implemented by NA-SA through upgrading measures. Once properly completed, these items will be seismically qualified and will fulfil the screening criteria. Many have already been solved and others are in process.
- Category 2 corresponds to items dependent on a relay/contactor chatter evaluation. A contract was agreed with INVAP to do a test of the relays and verify their qualification for the seismic review.
- Category 3 corresponds to items dependent on both easy fixes and a relay chatter evaluation (both are in process).
- Category 4 corresponds to items for which HCLPF calculations were performed. Ten (10) HCLPF capacity calculations were performed under Phase 4, covering 17 items classified as outliers.
- Category 5 corresponds to items to be evaluated in Phase 5 of the seismic safety evaluation programme (under process). They correspond to the following:
  - Structural capacity and safety margin of the Second Heat Sink Building;
  - Masonry Walls (interaction effects);
  - Refuelling Machine and Tilting Bottle;
  - NSSS seismic capacity evaluation;
  - Components of the Moderator System (MOV FIAT valves, heat exchangers, vertical pumps).
- Category 6 corresponds to items dependent on NA-SA decisions concerning the replacement of a number of them, such as the existing original diesel generators of the emergency power supply system.

The SHS system belonging to CNA I NPP is capable of removing heat in situations where the pump house is unavailable. The breakage of the above mentioned dams was considered in the CNA II NPP design basis so that its pump house is designed considering the resulting extreme water level. The CNA I NPP added a fourth pump to its secured river water cooling system (UK) which is housed in the CNA II pump house, capable of withstanding the maximum river level. In turn, systems SHS and UK were reviewed because they are included in the safe shutdown equipment list (SSEL) of the plants.

Containment isolation valves, fuelling machine, fuel storage building, cooling equipment and the fire fighting system were included in the SSEL.

As a result of the mentioned assessments, it was decided to implement the following improvements:

<u>Easy fixes:</u>

The plant walkdown was used to identify easy fixes as well as emergency measures aimed at reducing the vulnerabilities due to problems of interaction (impact, drop, spray, flooding, etc.), and other internal risks related with temporary equipment. New easy fixes were identified during the second walkdown carried out during 2015. Among others, the following easy fixes have already been implemented:

- Electrical and I&C cabinets in raised floor: an anchor was implemented to the bottom and/or upper floor slab, reinforced in two horizontal directions.
- Batteries: additional restrictions were installed to the racks to prevent slippage.
- Panels of the control room: the panels were reinforced from above to the concrete wall behind.

<u>Seismic instrumentation:</u>

New seismic instrumentation was installed in both units of CNA. The following tasks were performed:

- Laying cables.
- Installing the annunciator panel in the control room.
- Four seismic sensors on the ground (2 in Reactor Building, 1 in Reactor Service Area, 1 in Regulations Room).
- An annunciator panel was also installed in the auxiliary control room of CNA II.

- A field sensor was installed near the meteorological tower.
- The equipment was started up by setting the specific software GEODAS.

Seismically induced internal flooding:

The possible seismically induced internal flooding was analysed during the systems walk downs.

CNA II

The SSA consists of the same five phases showed for the CNA I case. Phase 1, was carried out in October 2011. Phases 2 to 4 were completed in June 2013. Phase 5 was completed during 2015.

As a result of the mentioned assessments, it was decided to implement the following improvements:

Plant walkdown:

A first walkdown was carried out during June 2013, resulting in the identification of all the outliers in the SSEL, considering issues like systems interaction, construction and installation, maintenance and housekeeping, as well as the relevant modifications to be carried out. During April 2015 a new walkdown took place to observe those components that could not be inspected during 2013 walkdown because they were not yet mounted, and to walk through components qualified as outliers. As a result of this last walkdown, some components remained classified as outliers. By applying what was implemented for Unit I, the results of the walkdown were presented in the 6 categories previously mentioned. Some easy fixes have already been solved and others will be implemented in the next scheduled outage in 2017.

The results of the stress tests generated by the NPP's Licensee were included in a Stress Test Report, which was sent to the Regulatory Body, who assessed its content. The assessment result was the following:

- The Regulatory Body agrees with the approach followed by the Licensee for the SSA re-evaluation of the above mentioned plants, and considers that:
  - No significant weaknesses have been identified that require urgent actions.
  - The Licensee complies with both the design bases and licensing bases.
  - The SMA methodology is an appropriate approach and sufficient, considering the seismic hazard of the sites in question and the ageing of the plant.
  - With the SMA methodology the non-existence of cliff edge effects was confirmed.
- The evaluation of safety margins is consistent with the defence in depth principle, since the safety functions cover the level 3 for a seismic demand corresponding to an earthquake with a 10,000 years recurrence period and, the confinement function is verified at level 4 for a seismic demand corresponding to an earthquake with a higher recurrence period.
- Internal and external flooding caused by earthquakes have been analysed and it is considered that the Licensee is carrying out the appropriate actions to successfully meet these scenarios.
- For the purpose of increasing the capacity to respond against extreme external conditions, the Licensee proposes to implement a set of improvements that are acceptable.

Besides, as a Regulatory Body proactive decisions it was decided to perform, through the Instituto de Investigaciones Antisísmica "Ing. Aldo Bruschi" (IDIA) of the Universidad Nacional de San Juan (UNSJ), an independent updated assessment of the seismic hazard.

The Regulatory Body requested the realization of a comparison between the IDIA study and the corresponding one addressed by the licensee. The conclusion was that both studies have similar results and they constitute the basis for the SSA of CNA I and CNA II.

Based on this evaluation the Regulatory Body considers acceptable the RLE of 0.1 g PGA and decided that it must be used at zero level of free field for the SMA.

3.17.2.3.2.1.2. Flooding / low water level

The estimated maximum water height that would be reached on the Atucha site after the rupture of the Yacyretá dam located 1,200 kilometres upstream is similar to the one calculated for the simultaneous confluence of the two main river tributaries in case of a maximum precipitation (intense rains). For the above mentioned dam rupture, it was estimated that the probable maximum high water level (PMH) for the Atucha site is 8.45 meters. The PMH is not related to an exceeding frequency. It is calculated in a deterministic way, assuming the total rupture of Yacyretá dam, maximizing all adverse factors associated with a flood hydrodynamic model, considering that they occur simultaneously.

Moreover, a high water level of such magnitude can be predicted three to four months in advance and was estimated that would take about thirty days to reach the Atucha site.

The CNA I and CNA II main buildings were built on a 23 meters height plateau. Therefore, high water levels are not expected to affect these buildings since those are all placed at a level that provides a substantial margin from the PMH.

However, in the CNA I case, the water intake for the normal river water cooling system pumps and the secured river water cooling system pumps are located at a level of 6 meters and may be vulnerable to river level rise for it is less than the PMH of 8.45 meters. Besides, the minimum water level height for which the plant can operate is -1 meter given by the pumps water intake level.

To overcome these CNA I extreme high and low water levels design weaknesses, a fourth pump of the secured river water cooling system (UK) was installed in the CNA II pumps house. This pump keeps running even with a river level rise of 8.45 meters or low river levels of -2.00 meters. This allows the pump operation even in case the water level exceeds the CNA I pump house height or the water level is below the pump intake level.

CNA II has three pump houses. The pump house of the service water intake (UPD) contains two pumps of the secured water system PE. The other two pump houses, circulation water intake of the condenser (UPC) and service intake (UQB). The houses UQB and UPD are designed to withstand a PMH 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 1,000 years.

In relation to the spent fuel storage pools, from the structural point of view and based on what was above mentioned, it is estimated as unlikely that they would be affected by the Paraná river's high or low levels. From the functional point of view, considering the UK system loss, the pools would lose the cooling water. To deal with this scenario, a strategy for refilling the pools through a water supply system pump (UJ) specially installed to draw water from the groundwater was implemented. The corresponding guideline within the framework of Severe Accident Management Programme was drawn up.

New hydrologic and hydraulic studies, in order to supplement, update, and also make a prospecting considering future possible scenarios *were performed and finished in October 2016. The target was to reassess both high water level (flooding) and low water level considered in the design basis, taking into account the combination of the Paraná River tributaries maximum flow, the break of dams located upstream as well as the boundary condition of the entrance of Paraná River given by the Río de la Plata river levels.*

The maximum and minimum water levels in Atucha I-II site *were* reassessed. The following scenarios *were* considered:

1) For maximum level rise: a chain break of Itaipú and Yacyretá, in simultaneous with a maximum Paraguay river flow, maximum rainfall on Paraná river basin and extreme rise of Río de La Plata river. In all cases historical records were taken as reference. *For extreme rise of Río de La Plata River a 1,000 years recurrence was taken, considering an event duration of 24 and 48 hs.*

2) For minimum water level: a minimum Paraná flow river with a 100 years recurrence in simultaneous with the minimum historical level in Paraná de Las Palmas discharge point.

The following studies were completed in a first step:

- General description of the area of study.
- Weather and meteorological events in the area where Atucha site is located.
- Hydrodynamic Model.

Additionally, a hypothetical scenario involving an earthquake strong enough to break down Yacyretá Dam and causing, at the same time, a seismic movement at Atucha II was considered.

An earthquake of such magnitude would be detected by Atucha instruments and a number of components listed in the Safe Shutdown Equipment List will be checked.

On the other hand, the displacement of the maximum possible flood from Yacyretá Dam zone up to Atucha site would take about 30 days.

For CNA Unit I the fourth pump of the secured river water cooling system would remain available, so one circuit of the shutdown core cooling system would remain available. For CNA Unit II, the secured river water cooling system would remain available. These systems and components have been evaluated for earthquake RLE review.

CNA I and CNA II have three specific flood and low water level management instructions. In case of an extreme scenario like the one described before, the NPP will be kept in a safe shutdown state.

Additionally, in CNA I, the replenishment of water inventory to the SGs through the Second Heat Sink (SHS) system, in case of complete loss of feed water and of the residual heat removal system, was implemented. This increases the reposition time of water inventory to the SGs independent of secured river water.

For CNA II, water supply to the SGs (and replenishment of the spent fuel pools) in the long term will be possible through the plant fire fighting system (SGA). Furthermore, some plant modifications are ongoing in order to allow the connection between the SGA system and the water supply system of the SGs. The following guidelines were drawn up within the framework of severe accident management:

- Guide A 11 "Lack of cooling in fuel element storage pools" which includes the replenishment of water through the demineralized water supply system (GHC) or through the fire fighting water system (SGA).
- Guide A 05 "Supply and Venting of Steam Generators".

In October 2016 the final report, "Hydrometric Extreme Analysis", was released. Taking into account the different scenarios postulated in previous paragraph of this section, the analysis concluded:

1. *The probable maximum high water level (PMH) for the Atucha site is 8.45 meters, ratifying the criteria considered valid previously.*
2. *The minimum water level height corresponds to +0.00 m in MOP scale.*

3.17.2.3.2.1.3. Other external hazards: tornadoes; wind loads; lightning and intense rains

Other possible external sources of flooding different from Paraná River (prolonged local intense rains; lightning; the breaking of tanks belonging to other nearby plants, etc.) were analysed and it was concluded that there is no other source that can cause flooding.

Therefore, it was decided to re-evaluate the CNA I original conditions of impacts caused by objects thrown by tornadoes affecting the plant buildings.

The report elaborated gathers and evaluates national and international standards applicable to missile impact generated by tornadoes. A study was conducted about the physical characteristics of tornadoes, storms and missiles that could impact on the facilities; a probabilistic model on tornado's risk was developed and applied in order to determine exceedance curves and the return period of wind speeds caused by tornadoes and other severe storms in the site area. For this reassessment, a record of tornadoes spanning until 2013 was used. Finally, the general condition of each building of CNA I regarding safety against missile impact was detailed.

Regarding intense rains, lightning and tornadoes, the conclusion of the assessments was that a suitable margin exists and some conceivable weaknesses were identified as well as some improvements and modifications were proposed. The licensee decided to implement additional studies to confirm them.

3.17.2.3.2.2. CNE site re-evaluation

A list of external hazards for re-evaluation was prepared and they were prioritised in accordance with its safety impact on the original design, as follows:

- Earthquakes.
- Extreme meteorological phenomena (tornadoes and severe storms).
- Man-induced events (mainly explosions and fires, external to the plant site).

The following actions were carried out related to each external hazard considered:

- Collection and analysis of related documents and reports.
- Regarding the re-evaluation of the operational response in case of an earthquake occurrence, two regulatory requirements were issued in 1999, regarding the implementation of an updated program

of plant response to an earthquake occurrence. These requirements include a re-evaluation of the seismic safety within the framework of an integrated, systematic and updated program. Therefore, the plant accomplished the definition, procurement, installation and commissioning of new digital seismic instrumentation to detect seismic events, evaluating their severity and providing data to plant operators. The instrumentation, installed during the 2000 planned outage, allows the recording of the seismic activity and provides this information directly to the operator in the control room for decision making in case of the occurrence of a seismic event.

- Preparation and implementation of operating procedures (update and improvements) to assess plant physical damage and plant operational situation after an earthquake occurrence, and, thus, to help in the decision making process for continuing operation and long term plant safety assessment. It includes a number of inspections to be carried out to determine the status of the safety system and safety related systems and according to the inspection results, to determine the full power operation, hot shutdown, cold shutdown or plant start-up.
- Re-evaluation of tornadoes and severe storms hazard, including the energy transmission lines which are essential for the plant safety. A work plan was prepared, starting in December 1998. A complete database has been compiled adding the 20 years of additional data, since 1980, and a more refined model for assessing the tornado impact probability was used (i.e. Twisdale and Dunn).
- Concerning the re-evaluation of potential hazards from man-induced events at the plant site, the document of Design Basis Threat is periodically reviewed by NA-SA, and experts from the national organization in charge of the physical protection of the NPPs are continuously trained.

A revision of the risk analysis for external events and the assessment of the radiological environmental impact are going to be performed as a part of the CNE Periodic Safety Review. In particular, as it was mentioned before, a seismic re-evaluation is planned as part of the CNE Life Extension Project.

The analysis and assessments, as part of the stress tests carried out in 2011 / 2012, required by the Regulatory Body related with extreme external events (earthquakes; flooding / low level water; and others external events) were performed. The improvements proposed by the Licensee as the stress tests results include the following:

3.17.2.3.2.2.1. Earthquakes

The CNE was originally designed for a 0.15 g PGA DBE corresponding to an estimated recurrence period of 1,000 years. The Ground Response Spectra (GRS) adopted were Housner type.

In 1982, a seismic evaluation was performed for some typical mechanical and electrical components, which are part of the Safe Shutdown System. The GRS input for this evaluation was based on a Seismic Prevention National Institute (Instituto Nacional de Prevención Sísmica - INPRES) SL-2 level earthquake at a PGA of 0.35 g. The evaluation utilized different criteria from the original design criteria, such as different damping values. Some modifications were suggested in the report in order to enhance the seismic capacity of the plant.

In 1983, D'Appolonia performed a probabilistic seismic analysis based on the geologic and seismologic information available at that time. It was concluded that the IAEA Seismic Level 2 (SL-2) earthquake for the Embalse site was 0.26 g PGA (associated with a 7,000 years recurrence period). This earthquake level was used in the final verification of the structural design.

Based on the review of updated seismic hazard information and state-of-the art technology, an Embalse specific Uniform Hazard Spectra (UHS) corresponding to a non-exceedance probability level of 0.0001 per annum was developed in 2011, which reflects the *latest knowledge of the seismic risk at the site*.

Geological investigations were performed in order to fully characterize the seismotectonic setting of the Embalse region and quantify local fault activity. These investigations include the following activities:

- 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 soil surveys, 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. Samples of the soil will be gathered at the trenches, to be used in the next task.

- Age dating of fault movement, to determine the age of the last movement of the faults, as well as the frequency that the faults have moved in the past.

NA-SA performed a comprehensive review of the site seismic hazard and carried out a Seismic Margin Assessment based on PSA (PSA SMA). The SMA determines the seismic margin to resist earthquakes bigger than the DBE and it provides a measure of the plant's robustness to face a Review Level Earthquake (RLE).

The result of the probabilistic seismic hazard analysis (PSHA) finished in 2011, is the site seismic hazard, expressed as uniform hazard spectrum (UHS) corresponding to a 10,000 years recurrence period.

According to the regulatory frame, the above mentioned UHS was adopted as safety objective for severe core damage (SCD). Based on this UHS, the acceptance criterion is that SSCs for preventing SCD should have a HCLPF capacity of the least 0.39 g PGA, defined as the RLE. As a consequence, safety features will cover what was established in the defence in depth principle. The RLE was defined with a sufficient margin over the original DBE (0.15 g PGA), in order to improve the plant safety and find weaknesses that could limit their capacity to safely bear the consequences *of* a seismic event greater than the DBE.

Regulatory Body assessed the results of the stress tests and like for the Atucha site case, it agrees with the approach determined by the Licensee for the SSA re-evaluation and consider that:

- The evaluation of safety margins is consistent with the defence in depth concept.
- *Internal and external flooding caused by earthquakes have been analysed and it is considered that the Licensee is carrying out the appropriate actions to successfully meet these scenarios.*
- For the purpose of increasing the capacity to respond against extreme conditions, the Licensee proposes to implement a set of improvements that are acceptable.

The Regulatory Body with IDIA as technical support organization (TSO) has concluded a site seismic hazard review of the assessment conducted by the Licensee. The results allowed the Regulatory Body to define the following regulatory framework for CNE life extension project:

- New SSC´s designs must be done considering, as design basis, the load obtained from the updated UHS and;
- In relation to SMA, it was stated that in order to verify the reactivity control and core heat removal safety functions, a PGA of 0.39 g had to be adopted as RLE. For the associated SSCs the acceptance criteria is HCLPF ≥ 0.39 g.
- For containment functions a higher target was established, namely a HCLPF ≥ 0.47 g.

As a result of the seismic studies described above, it was required to implement the following improvements:

Emergency Core Cooling System (ECCS):

Based on walkdown observations and seismic capacity assessments, the capacity of some components of the high pressure and medium / low pressure stages of the ECCS were found to require upgrades.

During the refurbishment outage, the recommendations on the high-pressure stage were performed. Actions on the medium / low pressure stages will be addressed as part of a design modification to provide a bypass line for the ECCS heat exchanger.

Civil Structures:

A whole sector of the Service Building was reinforced to meet the seismic capacity objectives for containment functions. This sector houses containment isolation valves, the Secondary Control Area and the ECC pumps. Also an evacuation route from MCR to SCA was selected and seismically qualified. Block walls adjacent to equipment credited for safety functions were reinforced.

Other seismic upgrades:

The seismic walkdowns and capacity evaluations identified many other pieces of equipment that required upgrades. Most of these recommendations were implemented during the refurbishment outage, including:

- *Reinforcement of the supports of the Primary Heat Transport System heavy water head tank.*
- *Upgrades to the bridge columns and to the carriage of the fuelling machines.*
- *Addition of attachments between adjacent panels in MCR.*
- *A procedure to reset important relays that may have changed state as a result of the vibrations caused by the earthquake (relay chatter).*
- *Many other reinforcements of the supports of tanks, valves, pumps, panels, etc.*

A few items remain outstanding after the refurbishment outage, and are being duly addressed in terms agreed upon with the Regulator.

Spent fuel dry storage system (ASECQ):

The *canister* confining the irradiated fuel elements were verified by the constructor INVAP S.E. in 1991 using seismic site reevaluation PGA = 0.26g. This earthquake corresponded to SL-2 from IAEA Safety Guide 50-SG-S1 which was employed for the verification of plant structures.

The *canister* were re-verified in 2011 by AECL, using updated PGA and spectra, to confirm that they can withstand seismic loading in combination with other solicitations due to own weight and thermal load. Different states were modelled applying load to the silos in order to verify whether the combination of these loads can induce stresses endangering the integrity of both the containment and the shielding. The conclusion is that the combination of the new earthquake and, the own weight loads and the thermal load does not produce sufficient stresses as to endanger to the integrity of the shielding or the containment.

Operating procedure for post-earthquake actions:

An operating procedure supported by the SMS exists, which list all the activities to be performed after a seismic event, higher than the operating basis earthquake (OBE), has occurred. The procedure includes criteria to shutdown and to return to service the reactor, and inspection instructions.

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 calandria vault inspection port to provide additional pressure relief to maintain the calandria vault integrity following a severe core damage accident. It is foreseen to be implemented by the end of 2017.

3.17.2.3.2.2.2. Flooding / low water level

The CNE is located at the Embalse site in the Córdoba Province, on the south shore of the Embalse Lake, which is formed by a dam on the Río Tercero River located downstream of the plant.

The CNE external flooding due to an Embalse Lake water level increase above the spillway level is not possible because this level corresponds to 657.5 meters and the CNE ground floor is at 665 meters (7.5 meters more). In addition, based on the existing historical background data, it is known that the lake level never exceeded 2.0 meters over the spillway level very low compare with 7.5 meters providing an adequate protection margin against flooding. Based on the above, no additional measures to protect the plant against an external flooding are considered necessary.

The water *consumption of the lake is far from the coast at the level of the bed of the Río Tercero,* which is the main tributary to the lake. Historical *reference* data indicates that this level was never reached. Therefore, it was estimated that a low lake water level that could affect the CNE safety functions only could be possible in *the event of a rupture of the dam caused by a wave that exceeds it.*

Regarding internal flooding it have been analysed the probable leaks occurrence from some relevant pipes that could affect safety related SSCs located in the turbine building or in the service building. The analyses allowed the identification of SSCs involving, among others, main *feedwater* pumps, low pressure emergency core cooling pumps, compressed air for instruments and some coolers. It was estimated that in case the main *feedwater* pumps go out of service, the primary heat transport system (PHTS) is *capable to be* cooled. The emergency core cooling system flooding would not be a problem, because the PHTS cooling would be completed before its integrity is compromised.

The corresponding studies and improvements proposed by the Licensee in relation with flooding / low water level are the following:

- *A study was performed, with the objective of determining the stability condition of the Río Tercero dam and the consequences of different combinations of dams breaks and weather conditions on CNE.*
- An air compressor driven by a DG located in the turbine building has being installed in a higher level (100 meters). This is connected by valves to the air supply tanks of instruments which are necessary for the operation actions required for the reactor cooling.
- The secondary control room gateway located in the turbine building *was* reinforced.
- *Isolation and sealing of the centre of the plant lighting system in order to withstand the push of an internal flooding, that could be caused by an eventual water leak from the condenser, and installation of a main gate to, if necessary, make equipment changes. This improvement has been already implemented.*

The Regulatory Body carried out the assessment of the stress tests results and concluded the following:

- The need for some regulatory actions but not relevant weaknesses that require urgent actions.
- It has been verified the Licensee complies with both the design and licensing basis.
- For the purpose of increasing the capacity to respond to extreme situations the Licensee proposes to implement a set of improvements including the corresponding implementation schedules, which were considered acceptable.
- The consideration of flooding / low-water-level for Argentine NPPs is consistent with both domestic regulations and international criteria established at the design time. However, further studies were considered necessary to require for the Embalse site. For CNE, it *was carried* out a re-evaluation of the consequences of the occurrence of earthquakes on the existing dam located downstream of the plant.
- Internal and external flooding situations have been analysed. Regarding this issue it is considered that the Licensee is carrying out appropriate actions to successfully cope with these scenarios.

A study evaluating consequences of dam collapse was developed by Evarsa. It included a stability analysis of Embalse Río Tercero Dam, and a mathematical hydrodynamic simulation of Cerro Pelado, Arroyo Corto and Embalse Río Tercero Dam rupture.

The study concluded that Río Tercero Dam is stable under any natural condition, and it could be damaged only in case of a malevolent action. Several scenarios comprising combinations of ruptures of upstream dams (Cerro Pelado and Arroyo Corto) cause different flood levels in the plant. The worst of them is the simultaneous rupture of Cerro Pelado and Arroyo Corto in combination with a river level rise corresponding to 1,000 years recurrence period. This situation would cause a flooding of about 1 m high and a permanence of 4 hours.

3.17.2.3.2.2.3. Other External Events: Tornadoes; Wind loads; Lightning and Intense Rains

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. 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.

A re-evaluation of the risk of tornadoes for the Embalse site was completed by 2018. This re-evaluation included the response analysis and the existing margin for the safety related buildings and SSCs facing tornadoes. The effect of missiles caused by tornadoes was included.

A recent plant outage caused by loose material accumulation in the plant water intake occurred revealing a weakness existence. This loose material came from extreme drought and ground removal which affected large areas surrounding the plant as well as the action of intense rains and strong winds which swept this material. The measures taken as a consequence of this event are:

- Hiring divers to make periodic cleaning of the plant water inlet.
- Take steps to restrict planting and grazing livestock on land adjacent to CNE as well as the weeding or clearing of these areas.

Based on the assessments carried out, the Regulatory Body concluded the following:

- The need for some regulatory actions, but not relevant weaknesses that require urgent actions, was identified.
- It has been verified that the Licensee complies with both the design and licensing basis.
- The consideration of tornadoes and wind loads for Argentine NPPs is consistent with both domestic regulations and international criteria, established at the design time. However, it was considered necessary to require new studies re-evaluating the risk of tornadoes for the site, *which was already done. The result of this re-evaluation determines if it is necessary to require complementary actions.*
- Tornadoes, wind loads, lightning and intense rains have been analysed and it is considered that the licensee is carrying out the appropriate actions to successfully cope these scenarios.

3.17.3. IMPACT OF THE INSTALLATIONS ON INDIVIDUALS, SOCIETY AND ENVIRONMENT

The authorized limits to the environmental releases of the Argentinian NPPs under operation were set by the ARN for relevant radionuclides. The radiological protection criteria used by ARN to control the dose received by workers are consistent with the latest ICRP recommendations (see article 3.15.). The authorized limits to the environmental releases have not been modified with respect to those reported in previous National Reports.

The Argentinian NPPs operator disposes of Operating Procedures for Abnormal Events (POEAs) foreseen to manage abnormal events and mitigate the corresponding consequences. New POEAs have been issued as a result of the assessment developed in the stress tests (for more detail, see Article 14).

The development of Severe Accident Management Program (SAMP) was continued at CNA I-II site. As part of it, a Severe Accident Management Manual was issued in August 2015.

The following are related to the program:

- PI-A-01: “Severe Accident Management Organization Program”
- PI-A-02: “Control, file and distribution of SAMP documentation”
- PI-A-03: “Technical Adequacy of SAMP”
- AG 01: “A Guides Use Rules”

In the case of CNE the SAMP 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.

In case of a radiological emergency, the actions to be taken inside the NPPs, by all the intervening organizations, are considered in the application of each Emergency Plan (see article 3.16.). Emergency Plans *were updated for the new cycle of operation of the Plant. The changes are in line with the IAEA GS-R-part 7 document.*

As a result of latest WANO Peer Reviews a set of several important actions was developed at Atucha site and CNE. A new Emergency Response Organization was set, and new procedures were issued, such as the procedure regarding control and inspection of emergency equipment.

For CNA I-II, a new Internal Emergency Control Centre (CICE) was built and future actions are in progress in order to build a new External Emergency Control Centre.

For CNE, it is planned to set up a Municipal Emergency Control Centre far from the planning zone.

The Environmental Management System (EMS) implemented by the Licensee adopts the requirements of ISO 14001:2004 “Environmental Management Systems-Requirements with guidance for use”. The Management Manual and the procedures that complement it, provides the corresponding information and describes the EMS.

The Licensee Environmental Policy is developed, implemented, conducted, reviewed and maintained through the EMS. The EMS is applicable to the activities, processes, products and services that interact with the environment and that NA-SA may control and over which it has influence. The EMS scope includes:

- Power generation in CNA I-II.
- Power generation and Cobalt 60 production in CNE.
- Activities related to the Project of Life Extension of CNE.
- Activities related to new Nuclear Industry Project.
- Technical and administrative direction in its headquarters.

The Licensee Environmental Policy and its objectives are:

- Prevent pollution by making a continued effort and minimize the adverse environmental impact from its activities as well as operate nuclear generating facilities by using energy and natural resources in a rational way.
- Adaptation continued of the environmental management to the applicable regulations.
- Evaluate the potential risks of new projects and minimize the environmental impacts during its implementation.
- Encourage internal and external communication.
- Communicate the Environmental Policy to all staff and make it available to interested parties upon request.
- Inform customers and the general public about the benefits of the nuclear option and its contribution to environmental preservation.
- Promote the training of personnel in the care of the environment.
- Continuously improve the environmental performance.

3.17.3.1. Pre-operational Monitoring Programmes

The conceptual model of radionuclide dispersion in the environment, based on site-specific data obtained during the siting studies, can be used to develop a pre-operational radiological environmental monitoring programme.

During the licensing process of CNA II, the licensee was required to carry out pre-operational monitoring, its main purpose being to collect baseline radionuclide activity concentration data, against which those results obtained from an operational radiological environmental monitoring programme would be later compared. It had been previously concluded that the same operational monitoring programme would be adequate for both NPPs within the Atucha site; therefore, sampled media and sampling points used for the pre-operational monitoring of CNA II were mostly selected based in the ongoing operational monitoring programme established for CNA I.

The radiological environmental baseline was developed between July 2012 and July 2013 with the assistance of Facultad Regional General Pacheco from Universidad Tecnológica Nacional (UTN).

The activities involved sampling, laboratory test and result analysis, and a contaminant transport study applying mathematical modelling in water and air.

It was studied superficial and underground water, soil, environmental dose rate, river sediments and milk.

ARN reviewed CNA II´s pre-operational monitoring programme and controlled its results using ARN´s own results from operational monitoring of CNA I.

3.17.4. COMPLIANCE WITH THE OBLIGATIONS IMPOSED BY THE CONVENTION

From the preceding considerations, it becomes evident that in the country the significant factors related to NPP sites that could affect its safety during their lifetime have been evaluated. Moreover, the radiological impact on the general public and environment due to their operation has also been evaluated. At the same time, the main site-related factors that ensure the continued safety acceptability of the NPPs have been re-evaluated or re-evaluation is being undertaken.

Therefore, the country complies with the obligations imposed by the Article 17 of the Convention on Nuclear Safety.

3.18. ARTICLE 18: DESIGN AND CONSTRUCTION

Each Contracting Party shall take the appropriate steps to ensure that:

i. The design and construction of a nuclear installation provides for several reliable levels and methods of protection (defence in depth) against the release of radioactive materials, with a view to preventing the occurrence of accidents and to mitigating their radiological consequences should they occur;

ii. The technologies incorporated in the design and construction of a nuclear installation are proven by experience or qualified by testing or analysis;

iii. The design of a nuclear installation allows for reliable, stable and easily manageable operation, with specific consideration of human factors and the man-machine interface.

3.18.1. INTRODUCTION

The Regulatory Body has issued standards that *covered* the necessary design and construction aspects in order to prevent accidents as well as to mitigate their radiological consequences if they occur.

These standards *were* compatible with deterministic concepts such as the defence in depth principle, and incorporate probabilistic concepts in order to define design criteria for the *existent* NPPs.

As it is mentioned in this National report, ARN is updating these standards in light of Fukushima's lessons learned as well as the Vienna Declaration, harmonizing the Argentine's safety standards with the corresponding IAEA.

For the new NPP's design *the regulatory expectations* are in accordance with the *latest concept of* defence in depth, *as stated in IAEA SSR2/1 Rev 1 and TECDOC 1791, "Considerations on the Application of the IAEA Safety Requirements for the Design of Nuclear Power Plants"* and in addition, comply with the criteria of redundancy, physical separation (segregation) and diversity specified by the Argentine Regulatory Standards.

Besides, requirements taking into account the prevention of SSC's eventual degradation *since early stage in the life cycle*, maintenance of safety systems reliability *targets*, and implementation of an emergency plan are included in the respective Operating Licenses. *Also, it is important to stress the continued safety improvement activities implemented in the existing NPPs through the development of regular PSR, in particular the safety factor design which yield the comparison of plant design against modern standards.*

3.18.2. DESIGN AND CONSTRUCTION

CNA I was designed before the defence in depth principle was first stated, but it also complies with the basic criteria associated with the principle. *For facing the LTO period, ARN requires that SSCs fulfil the engineering requirements (robustness, functional capacity and reliability) needed for having a robust Defence in Depth concept. Also, there must be in place design provisions for facing a broader design basis envelope by considering DECs.*

CNA II is conceptually similar to CNA I, but with more advanced safety aspects derived of the use of the Konvoi design and the operative experience gained from CNA I such as redundancy "2 out of 4" in relevant safety systems, better *construction measures like the use of* base material for the RPV *with high toughness and lower content of copper, nickel and manganese* and "stellite-6" elimination.

CNE was designed and built in such way as to count with levels and reliable protection methods against accidental release of radioactive materials (defence in depth principle), with the purpose of preventing accidents and mitigating their radiological consequences in case of occurrence. *During the refurbishment outage several safety upgrades in the plant design were implemented.*

The CAREM reactor, a project prototype of a small power NPP has *an enhanced Defence in Depth concept with* some distinctive and characteristic features that greatly simplify the design, and also contributes to a higher safety level, such as integrated self-pressurized primary cooling system, natural circulation, self-pressurized primary system and safety systems relying on passive features.

3.18.3. COMPLIANCE WITH ARGENTINE REGULATORY STANDARDS: GENERAL ASPECTS

Some Regulatory Standards (AR) were issued after the construction of CNA I and CNE, so the Regulatory Body did not ask for their immediate application. Nevertheless, CNA I and CNE design comparison against national and international modern standards as part of the Periodic Safety Review of these plants. The AR Standards have been applied for CNA II and CAREM at each stage of licensing.

The fuel elements are controlled, inspected, tested and verified according to the guidelines established in each installation's QA program, which comprises manufacture, transportation, reception and use stages.

The primary circuit integrity for both normal and design basis accidental conditions is preserved considering the effect of anchorages, connections, internal and external loads and deformations caused by thermal, mechanical and irradiation effects.

The NPPs have a containment representing the last confinement barrier, designed to support loads due to different accidental situations both from internal and external initiating events.

The containment system design criteria specially takes into account the number of confinement barriers, its retention capacity for radioactive material, its behaviour under normal and accidental loads, the leakage rate to the atmosphere and the results of the verification tests.

The shutdown systems (control rod drop and liquid poison injection) design criteria ensure the reactor shutdown in normal and accidental situations, keeping the safe state for the necessary time period.

The Emergency Electric Power Supply System design criteria allow the preservation of an adequate safety margin under normal and accidental conditions. It also complies with the independence, redundancy, physical separation and diversity criteria. External events such as fire and missiles are also considered.

Regarding CAREM reactor, considering that this is a prototype of an innovative design, the ARN has established an ad hoc licensing scheme, applicable to construction and commissioning stages.

3.18.3.1. CNA I NUCLEAR POWER PLANT

The reactor safety systems design and confinement barriers preventing fission product release, such as fuel element claddings, primary circuit and reactor containment, comply with the criteria established in AR Standard 3.2.1. Moreover, the safety systems design complies with the single failure as well as with the segregation and diversity criteria.

The core heat removal system design complies with the requirements of AR Standards 3.3.2. and AR 3.3.3. under normal operation (heat transport primary system and shutdown cooling system) and during hypothetical accidental situations (emergency core cooling system).

Both CNA I shutdown systems design comply with the criteria established in Standard AR 3.4.2., particularly as far as diversity, redundancy and reliability concerns.

The following systems constitute CNA I confinement barriers, as required by the Standard AR 3.4.3.:

- The containment system: this system is constituted by a steel sphere of approximately 50 m in diameter enclosed by a second safety cover of concrete, as its external shield. The system includes several penetrations, air locks and the isolation contention sub-system.
- Radioactive material removal system in case of an accident: this system is located between the steel sphere and the external shield and operates by passing contaminated air through carbon and absolute filters.

CNA I design also complies with the requirements of Standards AR 3.2.1., AR 3.3.1., AR 3.4.1. and AR 3.4.3., particularly regarding the uncertainty data boundary, and the application of safety concepts valid when its design was developed, such as redundancy, diversity, etc.

Such verification was obviously indirect, since the purchase contract specified that the SSCs, designed and manufactured in the Federal Republic of Germany, had to comply with the German standards because CNA I design had to be licensable in that country.

On the other hand, methods and calculation tools compatible with the state-of-the art of those times and verified through operation experience were used in the core design.

Regulatory Standard AR 3.2.1. criteria, related to the operator performance, are also fulfilled. The operator may always make provisions to avoid a situation that could affect the NPP's safety, but he must not avoid the necessary operation of safety systems. In any state of the NPP, all manually executed commands are subordinated to the reactor's protection system; therefore, reactor safety is not threatened by the non-detection of measurement devise readings or alarm signals, or any human error that could occur.

The Regulatory Standard AR 3.4.1., concerning man-machine interface taking into account the state-of-the art at the time the NPP was designed, regarding information processing and report systems is fulfilled. Particularly, during an appropriate time interval after the automatic activation of a safety system, no action is required by the operator who, on the other hand, is unable to avoid or interrupt its operation. Nevertheless, the operator may initiate other safety actions.

A new spent fuel element storage facility is being constructed because the available positions in the fuel storage pools were completed in October 2015 *(for more details* see section 3.19.9.3.1.).

Due to the new spent fuel storage facility has not been finished yet, a decision was taken to transfer spent fuel elements to the CNA II spent fuel pools. Devices and equipment were developed for this purpose. Argentine Nuclear Regulatory Authority authorized the transfer process of *1,477* CNA I fuel elements to CNA II pools. The fuel transfer requirements are average burned lower than 6,740 MWd / tU and minimum decay time of 33.5 years*. This process finished in May 2019, with 1,435 spent fuel elements transferred. Currently, a rearrangement of spent fuel elements between the pools is being carried out to allow access for those that will be transferred to the dry storage facility*.

3.18.3.1.1. Design improvements implementation

A stress test consisting in a reassessment of the CNA I´s safety margins assuming the occurrence of a sequential loss of the defence in depth caused by extreme initiating events, was carried out during 2011 / 2012. As stress tests results, the following design improvements have been or are being implemented to deal with beyond design basis accidents or to mitigate their radiological consequences:

- Installation of new seismic instrumentation.
- Installation of an additional (fourth) pump to the Secured River Water Cooling System (UK).
- Electrical manual interconnection between normal bars of CNA I / CNA II.
- Restoration of external power supply.
- Passive components control for spent fuel storage pool system.
- New Emergency Power Supply System (EPS).
- Alternative power sources (MDGs).
- Extension of the batteries availability. Disconnection of electrical loads.
- Dose rate´s remote measuring system.
- Alternative water sources: process water refilling of the spent fuel storage pools by using an alternative reservoir (with an independent pump) and replenishment of water inventory to the SGs through the SHS system.
- Implementation of WANO SOER 2011-2 recommendations.
- Installation of passive auto-catalytic recombiners (PARs).
- Increase in time of water inventory reposition to the SGs.
- Light water reposition to primary system through the pressure and inventory control system.
- Easy fixes coming from SSA program. Seismic capacity of the whole plant to cope with Design Basis Earthquake (DBE).

The following design measures / changes are still under assessment:

- Cooling of the RPV external side.
- Venting filtered containment system.
- I&C improvements to provide the information for severe accidents management (beyond design basis accidents). New level measurement was installed to manage water replenishment to the spent fuel pools.

Besides, in the frame of LTO, a comparison of the current design against the latest KTA standards is being carried out. From this comparison, design improvements to be implemented before the Phase B of the LTO period are identified.

One of the main improvements is related to the demonstration of the Break Preclusion Concept according to the KTA 3206. CNA I was designed neither consideration of the Break Preclusion nor Leak Before Break concepts in order to avoid the dynamic effects from the postulated failures. This demonstration has an important role in the identification of the protective measures from consequential failures needed in the plant according to current general layout.

Others foreseen improvements are related to the fulfilment of KTA 3501, "Reactor Protection System and monitoring Equipment of the Safety System" and KTA 3904, "Control Room, Remote Shutdown Station and Local Control Stations in Nuclear Power Plants".

3.18.3.2. CNA II NUCLEAR POWER PLANT

CNA II belongs to a second generation of PHWR type reactors with 745 MWe power installed. All the systems on the nuclear area were designed in a similar way to the German Konvoi PWR plants, except the heavy water specific systems (for details, see Annex III). The operative experience of CNA I was used, for example, to improve the reactor's internal design and the materials issued.

The Construction License was issued in 1981; and the construction process was delayed until 2006, when the Argentine Government decided to complete the construction and to proceed with the licensing process.

Bearing this in mind, the National Executive Power Decree No. 981/2005 instructed NA-SA, as Licensee, to conform the Atucha II Management Unit, whose objective was to carry out the activities which were required to put CNA II in operation.

On the other hand, the licensing activities for CNA II were carried out by ARN through the regulatory system described in article 3.7. of the present report. The regulatory activities consisted mainly in the update of the necessary features for the revalidation of the Construction License, the execution of independent safety assessments, and the inspection of assembly and tests carried out during the construction and commissioning stages and quality audits.

During the postponement period, the organization responsible for the construction worked on activities related to the maintenance of the already installed equipment and those stored, as well as in the documentation related to detail engineering and the update of the Safety Analysis Report (SAR). The SAR has been prepared following the US Regulatory Guide 1.70, Revision 3, and basically fulfils the recommendations of the IAEA Safety Guide No. 50-SG-G2.

In the meantime, ARN worked on the update and verification of the applicable ARN standards, the evaluation of the main safety aspects, the transference of operative experience from CNA I to CNA II´s design, quality audits carried out to the Licensee, etc.

Before making the decision of accelerating the finalization of CNA II works, the National Executive Power consulted ARN if the delays produced and the international advances in the state-of-the art related to this kind of installation could adversely affect the licensing process.

ARN analysed the conditions and came to the conclusion that:

1. There existed no restraint to continue the licensing process of CNA II, as far as the Licensee fulfilled the legal regulatory system in force that included specific requirements arisen from safety evaluations and inspections that would be performed in the future.
2. For the granting of the Commissioning License, the Licensee had to comply with the legal regulatory system according to Act No. 24,804 and its Regulatory Decree No. 1,390/98, meaning ARN's Regulatory Standards, regulatory requirements and sanctions, and with the international legal agreements according to what is established in the International Legislation accepted by the Argentine Republic on matters such as radiological and nuclear safety, security audits to nuclear material use, licensing and inspection of nuclear installations and international safeguards.

ARN based its opinion on the following issues:

1. CNA I is a second generation NPP whose design dates from the 60's and has been in operation since its commissioning in 1974 with a satisfactory safety level. The installation's operation for more than three decades, as well as the performance follow-up of other similar installations at

the international level, allows the acquisition of great operative experience that helped, in turn, to implement significant safety back-fitting improvements.

2. CNA II has much more advanced safety aspects than its predecessor CNA I, coming from the "Konvoi" design concept that was used.
3. Apart from these original design safety aspects, the operative experience of CNA I and the applicable international experience should be added, such as improvement of the reactor's internal design and the elimination of "stellite-6" in the core materials.
4. CNA II is the first NPP where the licensing process was made applying the Regulatory Standard AR 3.1.3. that considers in a probabilistic balanced manner the plant safety profile as well as the deterministic criteria normally taken into account.

ARN personnel with the assistance of domestic and foreign institutions carried out the regulatory tasks of evaluation, inspection and audits.

During the time that the construction of CNA II was postponed, one of the principal concerns was the appropriate components preservation. There was a preservation component programme in place which compromised routine and non-routine tasks.

The preservation processes were subjected to a continuous assessment by licensee internal and external quality audits, Siemens inspections, insurance company verifications and regulatory verification.

Additionally, an IAEA mission took place regarding the analysis of state of preservation of stored components and demonstration of fitness for continued use.

Personnel qualification, with the purpose of assuring an adequate process of preservation, was one of NA-SA's main concerns. Consequently the personnel that executed preservation tasks were trained and qualified according to NA-SA procedures, while Siemens/FANP qualified Preservation Supervisors and Preservation Teams.

The components preservation process results could be summarized as follows:

- Stored and erected items have been successfully preserved (including main components).
- Components and systems in operation have been maintained according to the maintenance program.
- A reduced quantity of non-critical items to be repaired or replaced has been identified.
- Criteria of specific revision of components and evaluation of possible replacement of parts subjected to natural ageing were applied during the pre-phase of the project.

The Regulatory Body performed independent inspections with the purpose of verifying the condition of the stored pipes and components before their installation in the plant. In order to apply this, ARN implemented a task force jointly with GRS´s experts. Visual inspections and non-destructive testing were carried out with satisfactory results.

3.18.3.2.1. Critical technical issues established by ARN

ENACE, the existing organization in 1981, responsible for the design and construction of CNA II (made up by 75% CNEA and 25% SIEMENS) was dissolved; therefore establishing a new organization to replace it became necessary.

As a result of additional evaluations of the SAR carried out by ARN during 2006, and taking into account the international state-of-the-art, ARN established, as first priority, a set of critical technical issues to be solved, by the Licensee providing adequate solutions. The critical technical issues were:

- Update the Quality System.
- Design Authority.
- Review of basic licensing criteria.
- Review of safety issues.

Update the Quality System: In order to comply with an ARN requirement, in 2007 NA-SA sent to ARN the QAP 115 - Rev. 3 "Quality Management System, Safety, Security and Health Program" for the Design, Construction, Commissioning and Operation stages of CNA II. This was performed taking into account the original QA program (Overall Quality Assurance Program QAP 115 and QSP 4a and 15a/c Rev. 2 mentioned in PSAR of CNA II).

The QAP 115 program ended when the Unit II reached 90% of full power. The QA Manual Rev. 5 for the site Atucha Unit I and Unit II was issued in 2014.

Design Authority: With the purpose of meeting this requirement and assuming the project direction, during the years 2004 and 2005 NA-SA reached an agreement with Siemens to provide supplies and services for the conventional area of the plant. Since then, AREVA provided experts for the project in engineering, licensing, erection and commissioning areas, AECL provided experts, engineering packages and supplies and IAEA sent two missions during 2006 and 2007 regarding the analysis of state of preservation of stored components and demonstration of fitness for continued use and sent experts with the objective of analysing and making recommendations about the situation of the Design Authority.

Besides, UNIPI developed a platform for thermal-hydraulic design issues and deterministic safety technology, the GRS assisted NA-SA with the necessary support to perform a PSA Level 2 for CNA II and CNEA was the strategic partnership to perform activities in different areas such as engineering, licensing, erection and commissioning.

Furthermore, CEN/SCK provided integral solution for the surveillance program of the RPV, EMC2 performed an evaluation of the break opening time for a 2A break LOCA and TECNATOM signed a contract for a full scope simulator for CNA II.

On the basis of the criteria set out in IAEA's document INSAG 19 "Maintaining the Design Integrity of Nuclear Installations throughout Their Operating Life" and taking into account the recommendations issued by IAEA experts during the above-mentioned mission, NA-SA as Responsible Entity distributed the responsibilities of the DA at three different levels of its organization in order to control the design changes:

The first instance of review is conducted by a Relevant Design Changes Review Committee (CRMRD) composed of external experts in various disciplines being their role to advise the highest authorities of the Primary Responsible of the operation.

The second review is done through the action of a Technical Review Committee (CRT) composed of specialized technical personnel.

Even though the CRT, normally devoted to the treatment of changes in operation NPPs, advises in nature to the General Manager of NA-SA, it becomes solvent in the case of relevant design changes of CNA II, with capacity to approve changes jointly with the General Manager.

Finally, the proposed changes relevant to safety are submitted for consideration and final approval to the Regulatory Body.

Another task to be fulfilled by the DA is the preservation of information. In this regard, NA-SA has defined that its Operating Unit, as Primary Responsible of the operation, be in charge for the determination of the engineering requirements, skills, and expertise required to comprehend the design of all SSCs important to safety.

Regarding the design integrity maintenance, a process of transference from the Construction Group to the Operation Group was carried out, including technical documentation and documented experience on construction and commissioning.

Review of the licensing basic criteria. By November 1977, the Regulatory Body signed with KWU enterprise the "Protocol of Understanding on the Basic Concept of Licensing and some safety aspects for Atucha II Project", which establishes that the Argentine regulatory framework does not use the deterministic concept of maximum credible accident. A probabilistic safety analysis including large LOCA was delivered to the Regulatory Body in April 2013 which has considered models´ improvement and new sequences with different break locations.

The Regulatory Body concluded that LOCA 2A must be considered as an accident that has to be covered by the safety systems, independently from its occurrence probability. In this sense, the Regulatory Body stated that, in spite the licensing basis was a 0.1A LOCA, larger LOCAs including the LOCA 2A must be considered deterministically as beyond design basis accident (BDBA)

Review of safety issues. CNA II is a PHWR so its original design involves a positive void reactivity coefficient. For that reason ARN considered important to analyse the installation's behaviour in those events that may lead to steam or void equivalent formation in the primary circuit, of which the LOCA can be considered as the most relevant.

The safety review was focused on the following activities:

a. Compliance with the regulatory criterion defined in Regulatory Standard AR 3.1.3.

b. Application of good international practices that arise from the following analysis:

   b1. Core design verification using modern coupled neutronic thermo-hydraulic models. Calculation performed based on adequate and validated codes.

   b2. Safety Systems verification (control rods, fast boron shutdown system, emergency core cooling system). Design Improvements of the fast shutdown system performance considering LOCA 2A as BDBA.

   b3. Fuel element behavior in case of LOCAs.

   b4. Update Break Preclusion demonstration from the original concept (3 lines) to the new one (6 lines) in order to deal with the consequential failures.

3.18.3.2.2. Design improvements implementation

A stress test consisting in a reassessment of the CNA II´s safety margins assuming the occurrence of a sequential loss of the lines of defence in depth caused by extreme initiating events was carried out during 2011 / 2012. As stress test results the following design improvements have been or are being implemented to deal with beyond design basis accidents or to mitigate their radiological consequences:

- Electrical interconnection between normal bars of CNA I / CNA II.
- Restoration of external power supply.
- Dose rate remote measurement system.
- Alternative water sources: alternative water reservoir that allows maintaining the water supply to heat removal through the SGs and cooling the spent fuel storage pool and; additional system to water replenishment of the spent fuel storage pools.
- Installation of passive auto-catalytic recombiners (PARs).
- Alternative cooling mode of the DGs.
- Increasing DGs autonomy.
- Switchgear building ventilation.
- Extension of the batteries availability.
- Easy fixes coming from SSA program. Seismic capacity of the whole plant to cope with Design Basis Earthquake (DBE).

As stress test results the following design measures / changes are in progress:

- Alternative power sources (MDGs).
- Cooling of the RPV external side.
- Venting filtered containment system.

The cooling of the RPV external side is still under development and analysis due to the complexity of the phenomena involved and the fact that Atucha reactors have unique Lower Plenum design characteristics. *For more information see Section 2.3.*

3.18.3.3. CNE NUCLEAR POWER PLANT

The reactor safety system design and the confinement barriers preventing fission product release, such as the fuel pellet itself, the fuel element clad, the heat transfer pressurised circuit and the reactor building, comply with the requirements of Regulatory Standards AR 3.3.2., AR 3.3.3. and A.R 3.3.4.

The core’s heat removal system design complies with the requirements of Standards AR 3.3.2. and AR 3.3.3. under normal operation (primary heat transport system and shutdown cooling system), and during accidental situations (emergency core cooling systems, high, medium and low pressure stages and emergency water supply system).

The confinement barrier required by Regulatory Standard AR 3.4.3 is constituted by the following systems:

- Containment system: this system is constituted by the building reactor structure, its penetrations, airlocks and isolation contention devices.
- Pressure suppression system: this system is constituted by the dousing system and the building air coolers.
- Fission product removal system: this system is constituted by the ventilation and the reactor building atmosphere steam recovering system.

CNE design complies with Regulatory Standards AR 3.2.1., AR 3.3.1. and AR 3.4.3. which defines requirements, regarding uncertainty data boundary, and the application of safety concepts valid when it was designed, such as redundancy, diversity, etc.

On the other hand, methods and calculation tools compatible with the state-of-the art in those times and verified through operation experience were used in the core design.

Regulatory Standard AR 3.2.1. criteria, related to the Licensee performance are fulfilled. Concerning the intervention in case of accidents, the Licensee must always make provisions to avoid a situation that could affect the NPP safety, but he should not avoid the necessary operation of safety systems. The operation of the low-pressure emergency cooling system can be interrupted, since it is designed to operate during long time intervals, and under certain circumstances it may be necessary to discontinue its operation.

Taking into account the state-of-the art regarding the information processing and report systems at the time the NPP was designed; Regulatory Standard AR 3.4.1 related to man-machine interface is fulfilled.

The Dry Storage of Irradiated Fuel Elements System is composed by a transference cell; transference flask and the silos field. Each silo accommodates nine canisters with 60 fuel elements per canister, with a total storage capacity of 540 fuel elements. Today the total number of qualified silos is *248.* The stored inventory at the end of *2018* was *128,520* fuel elements in *238* silos.

3.18.3.3.1. Design improvements implementation

As it was mentioned, a stress test consisting in a reassessment of the CNE´s safety margins assuming the occurrence of a sequential loss of the lines of defence in depth caused by extreme initiating events was carried out during 2011 / 2012. As stress test results the following design improvements have been or are being foreseen to be implemented to deal with beyond design basis accidents or to mitigate their radiological consequences. The identified improvements *were* implemented during the life extension of the CNE:

- External power supply protection devices.
- Upgrade of the EPS system.
- Replacement of the Class III DGs.
- Alternative power sources (MDGs).
- Extension of the batteries availability.
- Alternative water sources: installation of a facility to connect a fire truck from outside the building pool to replenish water; water replenishment of the spent fuel storage pool through a connection from outside the building pool; two mobile cisterns containing 25,000 litres of stored water each; water supply line to the calandria vault from outside the reactor building; connection through a hose line from a fire truck to the ECCS pipes to allow the water addition to the dousing tank; additional fire truck containing 17,000 litres of water and; facility to connect a fire-truck from outside the spent fuels storage pool building.
- Modifications to the emergency water supply system (EWS).
- Installation of passive auto-catalytic recombiners (PARs).
- Venting filtered containment system.
- I&C improvements to provide information for severe accident management (beyond design basis accidents).
- Improvement of the safety system trip parameters.

- Improvement of the ECCS´ reliability.
- Secondary control room gateway reinforcement.
- Installation of an air recirculation filters system in the secondary control room (SCR).
- Addition of a seismically qualified rupture disc to the calandria vault.
- Air compressor driven by a DG located in the turbine building.
- *Isolation and sealing of the centre of the plant lighting system in order to withstand the push of internal flooding, that could be caused by an eventual water leak from the condenser, and installation of a main gate to, if necessary, make equipment changes. This improvement has been already implemented.*
- Improvement of the seismic capacity of the whole plant to cope with the updated DBE.

3.18.3.4. CAREM REACTOR PROTOTYPE

The concept of CAREM that belongs to the very low or low power nuclear plants was put forward from the very beginning as an advanced designed reactor, being the precursor of innovative concepts as regards safety. CAREM is a light water reactor with new design solutions, which contributes to its high level of safety, being the followings its main innovative aspects:

- Integrated Primary System.
- Self-pressurization.
- Passive Safety Systems.

Even though both technical and engineering solutions associated to the NPP´s technology and, the innovative design characteristics are correctly verified during the design phase, it was considered convenient to construct a reactor prototype to validate its design, manufacturing, installation and operational aspects as well as verification of SSC´s reliability. A more detailed description may be found in Annex V.

Due to this fact, CNEA, as the primary agency to conduct nuclear science and technology developments in Argentine Republic, proposed to the National Government, to carry out the construction of the CAREM Prototype Reactor, by means of the construction of a CAREM NPP of 25 MWe. The reactor size was selected taking into consideration the following reasons:

- It is the minimum electric power output compatible with the need of recovering the operation and maintenance costs within the Argentine market values.
- It is a size that tries to minimize the initial investment needed, considering its very low power.
- It is a reactor that has good possibilities of being commercialized, without the need of modifications, since it is a good way to introduce nuclear energy in developing countries, as it has costs that are comparable to those of research reactors.
- It has a size that allows a relatively easy change of scale into a source of power that could allow the supply to isolated areas and to satisfy the requirements of the distributed generation (in the range from 25 to 120 MWe).

The CAREM project was started in Argentina some years ago, and the original objective was to study the possibility of meeting an existing need in the nuclear industry: that of very small and small reactors to facilitate the introduction of electric generation of nuclear origin to countries that need to give their first steps in this field and with which Argentina already has a collaboration history (although the CAREM project might also be a good choice for countries that need to increase their current electrical production a small fraction).

This CAREM project has enabled Argentina to make an incursion into the area of NPP design, assuring the availability of an updated technology in the short and mid-term. The design incorporates the technology acquired in the design and construction of modern research reactors and the operative experience in NPPs, making possible the implementation of advanced design solutions.

The CAREM concept was first introduced in the conference of small and medium reactors organized by the IAEA in Lima, Peru in March 1984. Since then, some of the design criteria of CAREM have been used by other designers, originating a new generation of reactors, where the CAREM is chronologically one of the first reactors with the greatest level of development considering the engineering experimental facilities constructed to validate design and codes.

CAREM-25 was presented for its analysis in several international forums, for example, between 2001 and 2002 the US-DOE (Department of Energy) and the Generation IV International Forum (USA), evaluated different technological alternatives of nuclear electric generation, including the CAREM. Argentina is one of the countries that integrate the above mentioned Forum. The CNEA is also active representing Argentina at the INPRO (International Project on Innovative Nuclear Reactors and Fuel Cycles), within the scope of the IAEA.

The above mentioned evaluation was in charge of approximately 100 experts of different countries, belonging to governmental organisms, universities and associations such as IAEA (International Atomic Energy Agency); AEA (Atomic Energy Agency –International); CNEA (Argentina); AECL (Atomic Energy of Canada Limited); Atomic Energy Commission of France; COGEMA (France); JAERI (Institute of Atomic Energy Investigations, Japan); KAERI (Institute of Atomic Energy Investigations, Korea); CNEN (Spain); ANL (Argonne National Laboratory, USA); DOE (Department of Energy - USA); EPRI (Electric Power Research Institute, USA); INEEL (Idaho National Engineering and Environmental Laboratory - USA); ORNL (Oak Ridge National Laboratory, USA) and Massachusetts Institute of Technology (USA), and supplier companies of nuclear plants, fuels or energy generators such as: BNFL (British Nuclear Fuels); Electricite de France; Framatome (France); Toshiba (Japan); Dominion (USA); Exelon (USA); General Atomics (USA); and Westinghouse (USA).

The results of that evaluation for the CAREM case were:

- It has an evolutionary design available for its construction in the short term.
- It is outstanding in terms of safety.
- From the economic point of view, it is above the average.
- The fuel utilization and the handling mechanisms are advanced.
- It is considered a feasible project and Argentina has the capacity of achieving it.

In this case, as well as in other evaluations, the results have been more than satisfactory.

3.18.3.4.1. Design and Construction of CAREM prototype

National Congress, through Law No. 26,566, 2009, declared of national interest the activities for the design, construction and commissioning of the CAREM prototype reactor to be built in Argentina, committing CNEA for that purpose.

CNEA, as the owner of the technology and Design Authority of the CAREM Reactor, is responsible for looking after safety matters as well as planning and construction solutions. CNEA provides scientific and technological backup to the works related to the project and is responsible for the proper implementation of tests and qualification of fuel elements and safety components.

In line with the integration of national capacities, the participation of domestic companies is preferred, such as INVAP S.E., Industrias Metalúrgicas Pescarmona S.A. - IMPSA, NA-SA, among others, which have the technological capacity to cover engineering and manufacturing of heavy components and contribute to the project implementation, are being selected.

CNEA has fostered three companies with great technical capacity in the nuclear area that will play an important role in the CAREM 25 Project. These companies are CONUAR, FAE and *INVAP*.

It is expected that its construction will be finalized by the end of *2022*, including the tests without fuel that will take more than a year to be completed.

3.18.3.4.2. Budget and human resources

The National Government has allocated a budget to CNEA to begin the CAREM Prototype construction tasks, and budget previsions were made until the finalization of the project.

As it was informed in the *Seventh* National Report, tasks were conducted in relation to project organization, Quality Assurance for the project and upgrade of the engineering documentation were consolidated. In addition, computer tools were incorporated for design and to facilitate the documentation management. Some of these tasks continued during the *following period*.

The Project organizational structure was completed and the personnel dedicated to this project are composed by 360 technical and support people with full time dedication and 150 more with a dedication above 50%. Training activities continues on-going to satisfy the needs of the Project. The incorporation of young people reduced the average age of the personnel dedicated to the CAREM Project.

3.18.3.4.3. Licensing

As it was mentioned in Section 2.12. of this National Report, taking into account that CAREM reactor was considered a prototype of innovative design; the ARN has established an ad hoc licensing scheme, applicable to the construction and commissioning stages. In this sense, the Regulatory Body has established a licensing scheme applicable to prototypes for new nuclear reactor designs. This scheme provides for licensing in successive stages, the first of which corresponds to obtaining the "Authorization for Use of Site and Construction" (AUSC). The requirements for obtaining such authorization consisted of the presentation of information on the following aspects:

- Environmental Radiological Impact.
- Waste Management.
- Radiological Emergency.
- Design Information.
- Quality management.
- Project Schedule.

In September 2013, after the evaluation of the information required for the first stage of licensing scheme of CAREM 25, the Board of Directors of ARN granted to the CNEA the "Authorization for Use of Site and Construction" (AUSC.)

The AUSC allowed CNEA to start construction works of the CAREM 25 project, except for the Nuclear Module (the part of reactor Containment which houses the RPV). The full start of construction was conditioned to accomplish a regulatory requirement in order to update CAREM 25 Design Report considering some changes in the layout or in civil structures of the plant.

In December 2014 ARN issued the Authorization for the Construction of Nuclear Module of CAREM 25.

Regarding the Environmental Impact, in September 2013 the OPDS (State Environmental Agency) issued the "Environmental Certificate" for the CAREM Project completing the authorizations in order to begin the construction.

Between 2016 and 2018, the development related to the civil works of the project, observed delays due to the change of contractor established by CNEA to carry out the work. ARN participated in the control of the process of transition to the new civil works contractor.

From 2018 and up to the present, ARN lead an inspection program at the site of the CAREM 25 reactor related with construction of civil structures (currently underway). This program is developed by ARN inspectors supported by experts belonging to some of the TSOs mentioned in National Report.

ARN reviews the mandatory documentation presented, in order to define and plan the inspection tasks (as were mentioned in National Report Section 3.7.3.3.

3.18.3.4.4. Design and Engineering Tasks

After finalizing an extended basic engineering, the detail and construction engineering of the safety and auxiliary systems, the reactor protection system and the control system are on-going including qualifications processes.

3.18.3.4.5. Construction Tasks

The civil works began after obtaining the necessaries authorizations and in February 2014, the first structural concrete was poured out. Nowadays the construction works are on-going.

The RPV is under construction with all the requirements to get the N stamp.

The *construction of the conventional building including the turbine, the generator and the tertiary system has already begun.*

3.18.3.5. FOURTH NPP

In November 2018, NA-SA and ARN have formally signed, as part of pre-licensing activity, a Memorandum of Understanding based on a general design's safety level evaluation oriented to the licensing feasibility of a fourth NPP. The fourth plant would be a HPR-1000 (Hualong I) designed by China National Nuclear Corporation (CNNC).

HPR-1000 is a Chinese advanced PWR nuclear power plant based on mature 3-loop PWR technology with operation experiences feedback into design characteristics. The design scheme adopts the advanced concept of GEN-III nuclear power plant and makes full use of successful experiences of Chinese PWR nuclear power plant design, construction, commissioning and operation, as well as it considers the latest research achievements.

HPR-1000 satisfies the requirements of currently nuclear safety codes and guides issued by the National Nuclear Safety Administration (NNSA - China) and the latest IAEA safety standards, like SSR 2/1 Rev.1. In addition, HPR-1000 satisfies the targets required by GEN-III utility requirement documents, integrated with safe and economic consideration.

The main design targets of HPR-1000 are as follows:

- *Unit rated power is ≥1200MWe (at 23ºC).*
- *Plant design lifetime is 60 years.*
- *Average Plant availability no less than 90%.*
- *Collective occupational exposure dose less than 1 man·Sv/reactor year.*
- *CDF $<10^{-6}$/ reactor year.*
- *LRF $<10^{-7}$/ reactor year.*

The main design features of HPR-1000 are as follows:

- *Active + passive safety design features.*
- *Complete severe accident prevention and mitigation measures.*
- *Single unit layout providing better physical separations for safety systems.*
- *Double-shell containment with enlarged free volume.*
- *Protection against large commercial aircraft crash.*
- *Improved configuration of safety injection system.*
- *Advanced fuel assembly with 18-month refuelling cycle.*
- *RPV Top-mounted advanced in-core instrumentation with on-line Linear Power Density (LPD) and Departure from Nucleate Boiling Ratio (DNBR) monitoring.*
- *Radioactive waste treatment.*
- *Enhanced emergency response capability.*

Despite that the contract between NA-SA and CNNC has not been signed yet, it is important to stress that a management system of the entire design process will be in place in order to assure that the design organization has the capability to provide design products and services complying with requisites set by NA-SA and applicable laws, regulations, standards, specifications and any other requirements established by ARN.

The Memorandum of Understanding will be part of the Licensing Basis Document where all codes and standards applicable to the whole project realization will be listed as of mandatory fulfilment.

3.18.4. COMPLIANCE WITH THE OBLIGATIONS IMPOSED BY THE CONVENTION

In Argentine, NPPs have been and are designed and constructed in such a way as to have several reliable protection levels, in order to prevent the release of radioactive materials to the environment, prevent accidents and mitigate their consequences in case they occur.

Therefore, the country complies with the obligations imposed in Article 18 of the Convention on Nuclear Safety.

3.19. ARTICLE 19: OPERATION

Each Contracting Party shall take the appropriate steps to ensure that:

i. The initial authorization to operate a nuclear installation is based upon an appropriate safety analysis and a commissioning programme demonstrating that the installation, as constructed, is consistent with design and safety requirements;

ii. Operational limits and conditions derived from the safety analysis, tests and operating experience are defined and revised as necessary for identifying safe boundaries for operation;

iii. Operation, maintenance, inspection and testing of a nuclear installation are conducted in accordance with approved procedures;

iv. Procedures are established for responding to anticipated operational occurrences and to accidents;

v. Necessary engineering and technical support in all safety-related fields is available throughout the lifetime of a nuclear installation;

vi. Incidents significant to safety are reported in a timely manner by the holder of the relevant licence to the ARN;

vii. Programmes to collect and analyse operating experience are established, the results obtained and the conclusions drawn are acted upon and that existing mechanisms are used to share important experience with international bodies and with other operating organizations and regulatory bodies;

viii. The generation of radioactive waste resulting from the operation of a nuclear installation is kept to the minimum practicable for the process concerned, both in activity and in volume, and any necessary treatment and storage of spent fuel and waste directly related to the operation and on the same site as that of the nuclear installation take into consideration conditioning and disposal.

3.19.1. INTRODUCTION

The ARN authorized NPPs commercial operation on the basis of the judgements mainly supported by both design safety assessments and commissioning follow-up results at the installations.

Once in operation, NPPs are operated by the Licensee according to what is established in the Operating License, as well as Mandatory Documentation i.e. Operational Limits and Conditions set in the SAR and the Policies and Principles Manual. This last document is based on the technical specifications of the installation and the operating experience.

ARN inspectors verify that the Mandatory Documentation that includes the three above mentioned documents are fulfilled. Furthermore, as part of routine inspections, resident inspectors audit and control procedure fulfilment, regular test performance, ISI, programmed maintenance and any other safety related activity.

The Reactor Manager is supported by an engineering section providing part of the technical support needed for the NPP operation. In addition, the Licensee also has an engineering division satisfying some of the technical support the installation needs. In order to cover other required services, domestic or international contractors are used.

The process, through which the installation operating experience feedback is carried out, both at the Reactor Manager and at the Licensee level, must comply with the Operating License requirements and Regulatory Standard AR 3.9.2. as well as with other applicable regulatory requirements.

The feedback process of operating experience of domestic NPPs involves the following entities: Licensee, ARN, Designers, Component Suppliers and international organizations dedicated to information distribution.

Furthermore, NPPs have programs for fire protection and management of radioactive wastes generated during their operation. The later program includes low and medium radioactive waste treatment and its subsequent storage.

3.19.2. INITIAL AUTHORIZATION TO OPERATE

3.19.2.1. CNA I

On May 31st, 1968, a contract between CNEA and Siemens was signed for the construction of CNA I. It was established that concerning radiological and nuclear safety, the design should comply with standards, rules and laws in-force in the Federal Republic of Germany.

With the purpose of carrying out both safety assessment and independent inspections, CNEA signed in 1969 a contract with the German Inspection Organization Technischer Überwachungs Verein, Baden (TÜV).

In 1971, the TÜV Baden issued a report concerning CNA I's construction, mainly containing a series of requirements, recommendations and additional information requests. It also carried out inspections to the manufacture of electric mechanical components assigned to CNA I. Later on, and during the electric and mechanical assembling stage, it designed a test and inspection plan for safety related systems. In 1972 the contract with TÜV Baden ceased, and the CNEA assumed the responsibility of carrying out the tests and inspection plan.

A commissioning ad-hoc committee called NPPs Safety Control and Inspection (CISIN) was then constituted within the CNEA, with the responsibility of evaluating and putting into practice requirements, recommendations and still pending additional information requests, as well as advising their authorities concerning CNA I and its operation and personnel licensing process.

3.19.2.2. CNE

CNE's initial authorization was issued according to the requirements established in Regulatory Standards AR 3.8.1. "Preliminary testing and commissioning of nuclear power plants" and AR 3.8.2. "Nuclear commissioning of nuclear power plants" (effective at that time). The first one established that the Licensee had to have a Pre-Commissioning Program and an Organization to carry it out. The prenuclear commissioning program comprised those tests required to demonstrate the safe operation of the NPP.

Regulatory Standard AR 3.8.2. also established that the Licensee had to have a nuclear commissioning program and an organization to carry it out. The standard also established that the Licensee had to appoint an ad-hoc committee for the nuclear commissioning follow-up, constituted by qualified personnel having experience in NPPs design, construction and operation. The ad-hoc committee had the main responsibility for evaluating each of the stages the commissioning program, and authorizes the transition from one stage to the other.

During pre-nuclear and nuclear commissioning stages, the ARN verified that the Licensee complied with the mentioned standards.

3.19.2.3. CNA II

The Commissioning of CNA II was divided into A, B, and C phases.

Phases A and B of the Commissioning Program were oriented to systems and are defined as Preliminary Tests in Argentinian Regulatory Standard AR 3.8.1., whereas Phase C was developed as a comprehensive unique Program for the whole Plant and is defined as Commissioning (meaning Nuclear Commissioning of the NPP).

Commissioning License was issued by the Regulatory Body allowing the nuclear commission of the unit, i.e., the first criticality and subsequent execution of an extensive testing program. The first criticality of CNA II occurred on June 3rd, 2014.

The nuclear commissioning program contemplated a gradual power increase in several power steps of 5% FP, 30% FP, 50% FP, 75% FP, and 100% FP, with the execution of the corresponding tests for each step, and subsequent Regulatory Body release of the next power increase after evaluation of the tests results.

Tests were carried out at power levels that are technically and physically the most expedient demonstration of proper functioning of the Plant.

All the 100% full power tests foreseen in the Phase C Program agreed with the Regulatory Body were completed successfully and the overall plant behaviour proved to be as expected according to the design basis and the safety analysis.

Operating License of CNA II was granted according to the requirements established in Regulatory Standards AR 3.8.1. and 3.7.1. after all the safety aspects of the plant were successfully verified. This includes completion of the objectives of the commissioning stage, and validation of the Mandatory Documentation and its approval by the Regulatory Body.

3.19.3. OPERATIONAL LIMITS AND CONDITIONS, MAINTENANCE, AND TESTING

The ARN in a continuous manner actively monitors the fulfilment of the operational limits and conditions of the NPPs, as well as the suitable development of maintenance programs and routinely tests through resident inspector activities carried out in each plant.

3.19.3.1. CNA I AND CNA II

The conditions for the authorization of the commercial operation both the CNA I and the CNA II were established in their respective Operating Licenses. The main requirements for both NPPs, such as maximum reactor thermal power, authorized discharge limits, communications to the ARN of the occurred significant events, etc. are explicitly contained in the License, or referred to in other mandatory documents.

The existing information related to operational limits and conditions of both CNA I and the CNA II are provided in their Policies and Principles Manuals that establish the maximum and minimum values for the operational parameters of the plants, ensuring compliance, during operation, of the situations considered in the design stage as well as the organization requirements that must be satisfied in order to ensure a safe operation.

The operational parameters concern mainly to reactor power, core reactivity control, heat transport systems, refuelling and secondary system related parameters. The specifications referred to the Licensee comprise, among others, safety related subjects, personnel licensing, minimum staff in plant and control room (see Regulatory Standard AR 3.9.1.), the Safety Advisory Internal Committee activities and the communication of significant events to the ARN.

CNA I and CNA II have a preventive maintenance program that aims to ensure process reliability of structures, systems and components of the plant, which include scope, planning, implementation and control of the preventive and predictive maintenance activities. All these activities are performed according to a set of procedures and manuals that are part of the mandatory documentation required in the Operating License.

Meanwhile, CNA I *and CNA II* developed the ISI *programs (in service inspection).* The *programs* include ISI activities related to significant components, equipment, and systems. *These programs are* routinely carried out, mainly involving the reactor pressure vessel (RPV); primary, moderator and volume regulation systems, as well as steam generators (SGs) tubes. Additionally, a CNA I ISI Manual upgrade was performed and an important review of the Periodic Test Procedures was carried out. Some more specifications were added, mainly related to test acceptance criteria.

Concerning CNA II, since its commissioning it has a preventive maintenance program that aims to ensure process reliability of structures, systems and components of the plant. All these activities are performed according to a set of procedures and manuals that are part of the mandatory documentation required in the Operating License.

Finally, the Internal Components Surveillance Program to CNA II is routinely carried out, involving mainly the inspection of the coolant channels.

3.19.3.2. CNE

The conditions for the initial authorization of commercial operation of CNE have been mainly established in the Operating License, where the essential requirements for the installation operation such as maximum reactor thermal power, limits of authorized discharges, communication of the occurred significant events to the ARN, etc. are explicitly contained or refer to other related documents.

In addition, another regulatory requirement conditioning CNE's commercial operation was issued including a set of complementary requirements to the operating license (see first Argentine report to CNS – 1998).

CNE *upgraded the* Policies and Principles Manual where operational limits and conditions for the safe operation of the installation are established *for the new life cycle*. Such operational limits and conditions mainly arise from the Canadian experience on CANDU type reactor operation, transferred to CNE. The Policies and Principles Manual is also the reference framework for most of the NPP's operational procedures.

During the execution of the life extension works (refurbishment activities), the operational status of CNE *was* framed in *a special revision of* the Manual of Policies & Principles.

CNE has preventive maintenance and ISI programs, which include scope, planning, implementation and control of the preventive, predictive and corrective maintenance activities. All these activities are performed according to a set of procedures and manuals that are part of the mandatory documentation required in the Operating License.

The surveillance programme includes ISI activities related to significant components, equipment, and systems. This program is routinely carried out, involving mainly the following: pressure tubes, primary heat transport, moderator and volume regulation systems, as well as steam generators.

As a consequence of a Regulatory requirement, a complete review of the Periodic Test Procedures was carried out. Some additional specifications were added, mainly related to test acceptance criteria.

3.19.4. INSPECTION

NPPs have Condition Based Maintenance approach which requires continuous monitoring and periodic inspection under preventive maintenance program definition, which include scope, planning, implementation and control of the preventive maintenance activities. All these activities are performed according to a set of procedures and manuals that are part of the mandatory documentation required in the Operating License.

Regarding inspection activities, the NPPs have in force surveillance program for long lived components, as the reactor pressure vessel and its internals, as well as in service inspection (ISI) for systems and components important to safety as primary, moderator and volume regulation systems, as well as steam generators tubes.

The ISI program in CNA II was issued and is being routinely executed. The Pre-ISI program was developed and accepted according to approved program and schedule. The results of the inspections were satisfactory and serve as a “Base-Line”.

3.19.5. OPERATIONAL PROCEDURES IN NORMAL AND ACCIDENTAL CONDITIONS

Most of the CNA I-II's operational procedures, either in normal or accidental conditions, are included in the Operating Manual.

The CNA I's Operating Manual has three parts:

- The first part has general plant descriptions, design parameters and operation mode.
- The second part has specific operation information; basically instructions to modify the installation's operational state, and instructions to perform infrequent handmade actions.
- The third part includes the manual of warnings and alarms of all the installation's boards, instructions for emergency cases and instructions for abnormal cases.

In case of CNA II, the Operating Manual contains the following parts:

- Part 1: Description of the Operations Manual.
- Part 2: Service Organization.
- Part 3: General Plant Operation.
- Part 4: Accidents and Disturbance.
- Part 5: Systems Operations.
- Part 6: Alarm Systems.

Most of the normal activities carried out at CNE are considered in procedures applied either in normal operation or accidental situations.

Procedures in Accidental Conditions are applied by different operation shifts during theoretical exercises as well as during simulator practices.

For a better arrangement of this report, all topics (including Procedures) related to the Severe Accident Management Program are included in Section 3.14.

3.19.6. ENGINEERING AND TECHNICAL SUPPORT

CNA and CNE NPPs have their own engineering sections. These sections are complemented by NA-SA headquarters technical services, which include specific subjects such as I&C and civil engineering and have a qualified staff of specialists who normally give support before and during the scheduled outages.

In some issues like non-destructive tests, materials, corrosion and water chemistry treatment, the NPPs Licensee requests the CNEA for service and specialised advice (technical support). Frequently they have also used the services of INVAP S.E. (an Argentine technology organization dedicated to advanced technology projects).

They have also used and will keep on using, if needed, the advice of foreign organizations such as Siemens - Kraftwerk Union AG responsible for the CNA I design and construction, Siemens and Areva, participants of the completion of building of CNA II, continuously providing technical support to CNA I and CNA II in both conventional and nuclear areas, and AECL responsible for the design and construction of CNE, as well as the companies that operate CANDU type reactors, with which there is an active experience exchange.

3.19.7. INCIDENTS REPORTING

One of the main concerns of the ARN is the occurrence of significant events and the actions related with them, considered as part of the profit gained from operating experience in NPPs.

Regarding the above mentioned, the Regulatory Standard AR 3.9.2. sets the basic criteria concerning definitions, event communication modes to ARN, and the events analysis by the Licensee. This analysis includes determination of root, immediate actions and corrective actions to prevent recurrence commensurate with the situations.

Furthermore, the Operating License sets particular conditions referred to the subject and some specific requirements have been issued concerning it.

During the period *March 2016 – March 2019, 49* relevant events were reported by *CNA I and CNA II*. The most significant operational events in CNA I, CNA II and CNE and how the Licensee and the ARN acted are given in Annex V.

3.19.8. OPERATING EXPERIENCE

In order to improve operational safety in CNA I, CNA II and CNE, a periodic analysis of their operating experience, and, to a smaller extent, an assessment of other NPPs operating experience are carried out.

As a consequence of a requirement issued by the ARN in 1998, the Licensee started a formal and systematic process of evaluation of the operating experience in order to obtain feedback to improve reliability and availability of the NPPs.

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 the 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 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).

The program prepared by the Licensee and presented to the ARN included goals to be reached, implementation procedures and the professional profiles of the working group staff. Emphasis was given to the improvement in safety of NPPs obtained from the feedback of operating experience.

To fulfil the 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. Besides, NPP's senior teams evaluate "low level events" and "near misses" creating their own database.

The result of the identification of direct and root causes of the selected events is transformed into corrective actions implemented in the NPPs, their effectiveness evaluated, and communicated to the others plants and to the ARN.

Whenever necessary, full event analysis using appropriate techniques such as Barrier Analysis and Change Analysis were performed. Also the use of "precursors" both from national and international Operating Experience have been used to avoid occurrence or recurrence of events.

The Regulatory Body has performed audits to the *CNA'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.

There are many improvement actions resulting from the feedback of the National Operating experience and Operating experience from Foreign NPPs in *CNA and* CNE. Many of them are being used in the backfitting program of the plants. Examples were shown in the previous National Reports.

The lessons learned from the events occurred in the domestic plants as well as the international operational experience are included in the general staff training program (annual safety course), and particularly in the training/retraining programs of both the managerial staff and the operational personnel, emphasizing the diffusion of the corrective actions arising from the events, to the plant personnel directly involved.

The ARN verifies that the Licensee addresses the lessons learned taking proper actions through a close follow-up of the actions taken by them. In addition, ARN verifies that such lessons are included in the corresponding training / retraining programs.

When applicable, the ARN promptly notifies the international community of the occurrence of a significant event together with its category according to the IAEA's International Nuclear Events Scale System (INES) and also informs the IAEA - IRS about the significant events occurred in the NPP, in order to enable the contribution of data about operating experience to other NPPs.

An event report is submitted to IRS when the event is considered by the national co-ordinator to be of international interest. Only events of safety significance are reported. The criteria to select the relevant events to be reported to the IRS are consistent with IRS User Manual guidelines. Among others, some of the most important guidelines are the following:

- Relevance of the event that will be shared from the point of view of the Operating Experience.
- Event Recurrence or misuse of the feedback from the International Operating Experience.
- Events related to design changes and component ageing.

As a result of the accident occurred at Fukushima Daiichi, the corresponding lessons learned were taken into account by both the Licensee and ARN. In this sense a stress test has been required by ARN to the Argentinean NPPs Licensee (including the NPPs in operation and under construction) consisting in a reassessment of the NPPs safety margin assuming the occurrence of a sequential loss of the lines of defence in depth caused by extreme external events in order to detect possible weaknesses and implement the corresponding improvements. Implementation of corrective actions for the findings was required to the Licensee.

Atucha Site has a common procedure to manage its Operating Experience. At present, CNA II Operating Experience, both internal and external, is managed by the same Operating Experience Division at Atucha Site, which has been formed with previous CNA I staff.

3.19.8.1. FEEDBACK FROM LOCAL OPERATING EXPERIENCE

CNA I, CNA II and CNE have, as part of their internal organization, an arrangement for the analysis of the operating experience, and carry out the resulting improvements and the information of results.

In CNA I, CNA II and CNE the following internal events are detected, recorded and analysed:

- Significant events, defined according to the criteria set in Regulatory Standard AR 3.9.2.
- Unforeseen outages.
- Minor events (low level events) or reportable events in CNA I, CNA II and CNE.

Although this task has particular characteristics for each NPP, the final results of the management of these events are similar. Each type of event is selected, analysed and if corresponds, the corrective action identified and implemented, and the information distributed in the NPP or in other NPPs according to specific procedures.

As regards significant events, the NPP procedures comply with the corresponding ARN Regulatory Standards AR 3.9.1. and AR 3.9.2. These standards establish criteria for the selection, analysis and information of the significant events occurred in an installation.

The criteria applied by the regulator and operator for screening other experience than incidents is mainly based on the lessons learned from domestic and international operative experience.

Concerning the screening of domestic experience, audits and eventually inspections results are used as a source of investigation related to management issues. For unexpected degradations, the results obtained from both the Ageing Program and the Surveillance Program (ISI and inspections) are used.

Concerning design weaknesses, the lesson learned from operative experience, deterministic and probabilistic safety assessments, as well as, a dynamic technical interaction with the designers are applied.

The external hazards considered are periodically reviewed according to the frequency revision as established in the mandatory documentation.

In CNE, as defined in the OPEX procedure, all personnel working at the plant, including contractors, have the obligation of reporting any “inappropriate condition” detected by issuing an inappropriate condition report (ICR). Those “Inappropriate Conditions” are screened daily by the Management who define if they need to be corrected and coded for trend “Finding / Minor Event” or declared as “Event / Minor Event” and analysed.

Most of the minor events and findings are not analysed as individual events. Instead these events are categorised, supplementing data is gathered and the data of the events are entered into the plant event data base. These events are analysed to identify any adverse trends. In the case an adverse trend is identified, an Apparent Cause Analysis is performed.

In a daily operational approach meeting, the ICRs reported in the last 24 hours are discussed aimed to that the Departments' Heads knows all the ICR awarded for execution. The ICRs can be considered as events or findings. When the Plant Manager, a Deputy Manager or a Department Head considers that an ICR must be treated as an event, the case must be presented to the mentioned daily meeting for consideration. In case that an ICR were categorized as a finding, the following actions are carried out:

- The corresponding plant sector is assigned responsible for implementing the corrective action.
- The above mentioned sector defines the implementation date of the corrective action.
- The finding is thus closed.
- OPEX Department assigns a code to the finding, performs the corresponding statistics and, includes it in an engineering report.

In CNA, all personnel working at the plant can originate the notification of a minor event. A committee constituted by members of the different sections (Operation, Engineering, Mechanical Maintenance, etc.) evaluates these events and proposes corrective actions. These actions are followed with the Action Corrective Programme. All the operational incidents, significant and minor events, their corrective actions and their follow-up are recorded in Operative experience Database.

In CNA I, CNA II and CNE every event implying an unforeseen outage and/or a deviation from operational limits and established conditions, must in addition be evaluated by the plant’s “Internal

Safety Advisory Committee" (CIAS) according to what is established in Regulatory Standard AR 3.9.1. Its conclusions and recommendations are written down in minutes signed by the CIAS members.

In addition, the Technical Revision Committee (CRT), independent from the installation, must analyse the importance of the foreseen operational incidents, and the occurred significant events. Its conclusions and recommendations are recorded in minutes signed by the CRT members.

A list of the events, lessons learned and corrective actions resulting from national and international operating experience in the reported period is detailed in Annex V.

3.19.8.2. FEEDBACK OF THE OPERATING EXPERIENCE FROM OTHER NPPs

At the beginning of CNA I's operation, its designer, Siemens - Kraftwerk Union AG, played an important role in the transmission of operating experience of the German PWR, applicable to that NPP. CNA II has also received the support of the original designer, Siemens-KWU, during the first years of construction. When CNA II construction was resumed in 2006, NA-SA took the role of Design Authority and has since continued the operating experience interchange with Areva, who is the owner of KWU designs.

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, Wolsong-I, in order to exchange operating experience. Moreover, it is member of the CANDU Owners Group since its creation.

Presently, both CNA (CNA I-II) and CNE NPPs receive information from the following databases:

- CANDU Owners Group (COG).
- World Association of Nuclear Operators (WANO).
- IAEA's International Reporting System.

The processing of the information provided by the different sources is managed by OPEX NA-SA Central Group. They look for the events, they analyse their applicability for each plant and distribute the external OPEX through a periodic newsletter. The most significant event included in the newsletter must be responded by CNA and CNE, the rest are considered for information, and distributed along the plant.

CNA I has been using the WANO database since 1996. The collection, selection and classification of information have been systematised.

CNE uses COG databases as part of its usual working activities. In addition, CNE usually participates in COG Weekly Screening Meeting teleconferences. Several corrective actions have been implemented as a consequence of the information received via COG. On the other hand, CNE provides COG a periodic report of its significant events, via the teleconferences mentioned above.

ARN examines the effectiveness of operating experience feedback using information coming from national and international databases.

This information is analysed by an analyst team 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.
- *Applicability analysis of external events.*
- Corrective actions.
- Corrective action follow-up.

The Licensee Headquarters Operating Experience Division performs a screening analysis using international databases selecting the applicable events for the domestic plants. This Division reviews more than 1,000 reports per year from different international sources, such as WANO, COG and IRS. However, due to design, procedures, systems and components features or low safety significant actions, approximately less than 10% of the events have a meaningful application for NA-SA NPPs.

The selected events, those which have a potential application, are presented in a quarterly newsletter to some of CNA, CNE and Headquarters staff members. The events are classified in two categories: "Significant" and "For information". The "Significant" events require a formal answer from NPPs. In CNA Site, the answer includes the *applicability* analysis for both plants. The Operating Experience

Division at each Site defines a responsible to answer the recommendations of the significant event. If the analyst at the plant finds a gap between the external event recommendations and the situation in his Plant or Site, he should issue a corrective action. If any of the "For information" external events is considered by the Plant to have useful information to implement, then the affected sector will issue a corrective action.

Since Fukushima accident in 2011, WANO has paid more attention to SOER recommendations implementation. Each Site has, as part of their Operating Experience procedures, a chapter for SOER management *with their specific procedure.* Each SOER has an "owner", designated by the Plant, who is responsible to revise the Site situation against the recommendations. If a gap is found, then a corrective action will be issued.

OPEX is one of the sources for design changes included on CNE planned refurbishment for life extension. Most of them were taken into consideration in previous periods, as CNE was in the Design and Procurement phases of the Life Extension Project. Nevertheless, there is still some useful Operating Experience that could be implemented during the Construction/ *Commissioning* Phase. *For the period 2016 - 2019, it could be mentioned the following:*

The Headquarters Operating Experience Division has selected *134* reports from international databases.

- The CNE Operating Experience Division has received *103* reports. Approximately *9%* of them are "Significant" and the rest are "For information".
- The CNE Operating Experience Division sent 59 responses to the events reported, including *8* proposals of corrective actions.

Examples of lessons learned from local and international operating experience in the period *2016 - 2019* are shown in Annex V.

3.19.8.3. PEER REVIEWS AND ACTIVITIES BETWEEN THE LICENSEE AND WANO FROM 2016 TO 2018

The Licensee is a WANO´s member (created in 1988), at the beginning through the CNEA and then its condition of associate as NA-SA was ratified in Paris (1995).

NA-SA received WANO Peer Review (PR) *and Corporate Peer Review (CPR) missions* and the corresponding Follow-Up (FU) - Atucha I NPP: PR in 2006 and 2012, FU in 2009 and 2015; Atucha II NPP: Pre Start Up PR in 2013, Return Visit in 2014; *Atucha Site: Crew Performance Observations (CPO) and Design Informed PR in 2017, FU in 2019; Embalse NPP: PR in 2007 and 2014, FU in 2009, CPO, Re Start Up PR, and Return Visit in 2018; NA-SA Headquarter: CPR in 2014 and CPR FU in 2018.*

NA-SA participates in the current WANO programs through the WANO PC (Paris Centre): Peer Review, Performance Analysis, Training and Development, Member Support and Corporate Communications. Argentina has also provided specialists from the NPPs to participate in every WANO Program.

All the activities during the reported period are listed in Table 19.1.

Table 19.1.

PERIOD	HOST	PARTICIPANTS	ACTIVITY
September 2014 - August 2016	*WANO Paris Centre France*	*NA-SA Corporate Engineering Management Engineering Applied to Processes Head of Section*	*WANO Paris Centre Secondee*
September 2014 - August 2016	*WANO Paris Centre France*	*NA-SA Corporate Engineering Management Electrical Equipment Head of Division*	*WANO Paris Centre Secondee*
September 2015 - August 2017	*WANO Paris Centre France*	*Embalse NPP Electrical Maintenance Engineer*	*WANO Paris Centre Secondee*

May 23 - June 10, 2016	*Torness NPP UK*	*Embalse NPP Instrumentation and Control Maintenance Head of Division*	*Peer Review (MA)*
June 2-3, 2016	*WANO Paris Centre - France (Teleconference)*	*NA-SA´s WANO Interface Officer*	*56th WANO Interface Officer Meeting*
June 21-22, 2016	*Liubliana Slovenia*	*NA-SA´s Governor*	*68thWANO Paris Centre Governing Board*
July 25-28, 2016	*WANO Atlanta Centre - USA*	*Embalse NPP Professional Instructor*	*Seminar on New Human Performance Lead*
July 10-11, 2016	*Washington USA*	*NA-SA´s CEO*	*Chief Executive Officers Meeting*
August 22- September 16, 2016	*Dungeness NPP UK*	*Atucha NPP Technical Assistance of Maintenance Manager*	*Peer Review (MA)*
September 2016 - August 2018	*WANO Paris Centre France*	*Atucha NPP Process Specialist*	*WANO Paris Centre Secondee*
September 23-30, 2016	*Angra NPP Brazil*	*Atucha NPP Engineering Head of Department*	*Benchmarking Visit on Modification Management*
September 23-30, 2016	*Angra NPP Brazil*	*Atucha NPP Modification Management Head of Division*	*Benchmarking Visit on Modification Management*
September 26, 2016	*WANO Paris Centre (Teleconference)*	*NA-SA´s WANO Interface Officer*	*NA-SA Member Performance*
October 10-28, 2016	*Beznau NPP Switzerland*	*Atucha NPP Shift Supervisor*	*Peer Review (OP)*
October 13, 2016	*NA-SA Headquarters*		*Annual Meeting*
October 18-20, 2016	*WANO Paris Centre France*	*NA-SA´s WANO Interface Officer*	*57th WANO Interface Officer Meeting*
October 24-28, 2016	*Vattenfall, Sweden*	*NA-SA Corporate Nuclear Safety and Licensing Manager*	*Benchmarking Visit on Independent Oversight*
October 24-28, 2016	*Vattenfall, Sweden*	*NA-SA Corporate Engineering Management Engineering Applied to Processes Head of Section*	*Benchmarking Visit on Independent Oversight*
October 25, 2016	*Helsinki-Finland*	*NA-SA´s Governor*	*69th WANO Paris Centre Governing Board Meeting*
October 25 - November 10, 2016	*Rovno NPP Ukraine*	*Atucha NPP Radioprotection Head of Division*	*Peer Review (RP)*
October 25 - November 10, 2016	*Rovno NPP Ukraine*	*Embalse NPP Training Department Analyst*	*Peer Review (TQ)*

November 21-25, 2016	*Embalse NPP*		*Peer Review Follow Up*
December 12-16, 2016	*Santa Maria de Garona NPP Spain*	*Embalse NPP*	*Technical Support Mission on Assurance the Adequate Addressed to the Risk Associated with Applicable Issues*
December 12-16, 2016	*Changjiang NPP China*	*Atucha NPP Shift Supervisor*	*Technical Support Mission on Leadership Development of (Deputy) Shift Supervisor and Conservative Decision-Making*
January 16-20, 2017	*Hong Yan He NPP China*	*Atucha NPP Mechanical Head of Division*	*Technical Support Mission on Leakage Rate Monitoring on Secondary and Third Barrier*
March 14-16, 2017	*WANO Paris Centre France*	*NA-SA´s WANO Interface Officer*	*58th WANO Interface Officers Meeting*
March 20-23, 2017	*NA-SA Headquarter*		*Technical Support Mission on Human Resources*
March 23, 2017	*WANO Paris Centre France*	*NA-SA´s Governor*	*70th WANO Paris Centre Governing Board*
April 17-May 5, 2017	*Tricastin NPP France*	*Embalse NPP Technical Assistant of Mechanical Engineering*	*Peer Review (EN)*
May 2017-April 2019	*WANO Paris Centre France*	*Atucha NPP Technical Assistance of Maintenance Manager*	*WANO Paris Centre Secondee*
May 15-19, 2017	*Atucha NPP*		*Crew Performance Observations*
May 29 - June 16, 2017	*Heysham NPP UK*	*Atucha NPP Nuclear Safety Technical Support Head of Section*	*Peer Review (EN)*
May 31 - June 16, 2017	*Ning De NPP China*	*Atucha NPP Safety Evaluations Head of Division*	*Peer Review (EP)*
June 7-8, 2017	*Leibstadt Switzerland*	*NA-SA´s WANO Interface Officer*	*59th WANO Interface Officers Meeting*
June 14, 2017	*WANO Paris Centre France*	*NA-SA´s Governor*	*71st WANO Paris Centre Governing Board*
June 26-29, 2017	*Vattenfall Corporate Office Sweden*	*NA-SA Corporate Human Resources Management Training and Human Factors Head of Section*	*Human Performance Working Group Meeting*

July 12-28, 2017	*Atucha NPP*		*Design Informed Peer Review*
July 24-28, 2017	*Neckarwestheim NPP Germany*	*NA-SA Nuclear Project Management Engineering Manager*	*Peer Review Exit Representative*
August 28 - September 15, 2017	*Hong Yan He NPP China*	*Atucha NPP Process Specialist Engineering*	*Peer Review (EN)*
September 25-27, 2017	*WANO Paris Centre France*	*Atucha NPP Control and Instrumentation Head of Department*	*Workshop on Scram Reduction Programme*
September 25-28, 2017	*Korea Hydro & Nuclear Power (KHNP) Korea*	*Embalse NPP Operating Experience Head of Section*	*Technical Support Mission on Root Cause Analysis Assistant Visit*
September 25-29, 2017	*Heysham NPP UK*	*NA-SA Corporate Enginnering Management Electrical Equipment Head of Division*	*Technical Support Mission on Plant Status Control*
September 26-28, 2017	*WANO Paris Centre France*	*NA-SA´s WANO Interface Officer*	*60th WANO Interface Officers Meeting*
October 5-20, 2017	*Smolensk NPP Russia*	*Atucha NPP Safety Assessment Head of Section*	*Peer Review (EN)*
October 9-27, 2017	*Trillo NPP Spain*	*Atucha NPP Human Performance Head of Department*	*Peer Review (OA)*
October 15-17, 2017	*Gyeongju Korea*	*NA-SA´s Governor*	*Biennial General Meeting and 72nd WANO Paris Centre Governing Board*
October 16-20, 2017	*Embalse NPP*		*Technical Support Mission on Operation Readiness Assistance*
October 16-20, 2017	*Tianwan NPP China*	*Atucha NPP Industrial Safety Head of Department*	*Technical Support Mission on Rigging, Lifting and Material Handling*
November 13-17, 2017	*Bohunice NPP Slovak Republic*	*NA-SA Corporate Management Techniques Head of Division*	*Technical Support Mission on Nuclear Safety*
November 20-December 8, 2017	*Civaux NPP France*	*Atucha NPP Fuel Management and Early Warning Head of Division*	*Peer Review (EN)*
December 2017-November 2019	*WANO Paris Centre France*	*Embalse NPP Technical Assistant of Mechanical Engineering*	*Secondee*

December 2-15, 2017	*Leningrad NPP Russia*	*NA-SA Corporate Quality Control Management Quality Control Head of Department*	*Pre Start Up (MA)*
December 5-8, 2017	*Simulator Centre Essen Germany*	*Atucha NPP Management of Human Performance Program Head of Division*	*Simulator Centre Essen Germany*
February 5-16, 2018	*Embalse NPP*		*Crew Performance Observations*
March 3-16, 2018	*Yangjiang NPP China*	*NA-SA Corporate Nuclear Safety and Licensing Management Technical Assisstant of Radiation Protection Deputy Manager*	*Pre Start Up Review (EP)*
March 6-8, 2018	*WANO Paris Centre France*	*NA-SA´s WANO Interface Officer*	*61st WANO Interface Officers Meeting*
March 14-16, 2018	*NA-SA Headquarter and Atucha NPP*		*WANO Representative Support Visit*
March 22-23, 2018	*WANO Paris Centre France*	*NA-SA´s Governor*	*73rd WANO Paris Centre Governing Board*
April 9-13, 2018	*NA-SA Headquarter*		*Corporate Peer Review Follow Up*
April 16-May 4, 2018	*Cofrentes NPP Spain*	*NA-SA Corporate Nuclear Safety and Licensing Management Licensing Management Head of Section*	*Peer Review (FP)*
May 7-8, 2018	*NA-SA Headquarter*		*Organizational Analysis Project Pilot Test*
May 28-June 8, 2018	*Embalse NPP*		*Restart Review*
May 28-June 15, 2018	*Belleville NPP France*	*Atucha NPP Senior Instructor*	*Peer Review (OP)*
May 30-31, 2018	*Cambrils Spain*	*NA-SA´s WANO Interface Officer*	*62nd WANO Interface Officers Meeting*
June 11-13, 2018	*Atucha NPP*		*Technical Specifications Support Visit*
July 3-4, 2018	*NA-SA Headquarter*		*Workshop on Leadership and Behavior Change Management*
July 7, 2018	*NA-SA Headquarter*		*Mid-Year Meeting*

August 30-September 14, 2018	*Kalinin NPP Russia*	*Atucha NPP Nuclear Safety Technical Support Head of Section*	*Peer Review (EN)*
September 3-7, 2018	*EDF Energy Corporate UK*	*NA-SA Corporate Quality Control Management Mechanical Quality Control Head of Section*	*Member Support Mission on Maintenance Quality Control*
September 12-October 5, 2018	*Hope Creek NPP USA*	*Electrical Systems Head of Division*	*Peer Review (EN)*
September 18-October 12, 2018	*Columbia NPP USA*	*Atucha NPP Electromechanical Head of Division*	*Peer Review (MA)*
September 24-28, 2018	*Bohunice NPP Slovakia*	*NA-SA Corporate Engineering Management Events Management Head of Section*	*Member Support Mission on External Operating Experience*
September 24-27, 2018	*WANO Paris Centre France*	*NA-SA´s WANO Interface Officer*	*63rd WANO Interface Officers Meeting*
October 2-4, 2018	*NA-SA Headquarter and Atucha NPP*		*International Human Performance Working Group Meeting*
October 8-26, 2018	*Vandellos NPP Spain*	*NA-SA Corporate Quality Control Management Process Improvement Analysis Head of Division*	*Peer Review (OP)*
October 11, 2018	*NA-SA Headquarter*		*Annual Meeting*
October 16-18, 2018	*CNAT Spain*	*NA-SA´s Governor*	*75th WANO Paris Centre Governing Board*
October 29-30, 2018	*Embalse NPP*		*WANO Representative Support Visit*
November 5-7, 2018	*Vandellos NPP Spain*	*Atucha NPP Operations Technical Office Head of Department*	*Benchmarking Visit on Technical Specifications*
November 5-7, 2018	*Vandellos NPP Spain*	*Embalse NPP Continue Improvement Head of Department*	*Benchmarking Visit on Technical Specifications*
November 5-7, 2018	*Vandellos NPP Spain*	*Atucha NPP Safety Assessment Head of Section*	*Benchmarking Visit on Technical Specifications*
November 12-16, 2018	*EDF Corporate Paris*	*Atucha NPP Human Performance Head of Department*	*Member Support Mission on Leaders in the Field Effectiveness.*
November 13-16, 2018	*Iberdrola Spain*	*NA-SA Corporate Nuclear Safety and Licensing Manager*	*Corporate Peer Review Follow Up*

November 22-December 6, 2018	*Narora NPP India*	*NA-SA Corporate Engineering Management Electrical Engineer*	*Peer Review (EN)*
November 26-December 14, 2018	*Bugey NPP France*	*Embalse NPP Instrumentation and Control Maintenance Technical Assistant Head of Division*	*Peer Review (MA)*
November 26-27, 2018	*WANO Paris Centre France*	*NA-SA Corporate Institutional Relations External Communication Head of Division*	*Communications Expert Group Meeting*
November 27-28, 2018	*WANO Paris Centre France*	*Atucha NPP Production Deputy Manager*	*Workshop on Improving Plant Performance*
December 3-7, 2018	*Oskarshamn NPP Sweden*	*Atucha NPP Process Specialist*	*Member Support Mission on System Health*
December 10-14, 2018	*Embalse NPP*		*Restart Review Return Visit*
January 2019-December 2020	*WANO Paris Centre France*	*Atucha NPP Senior Instructor*	*Secondee*
January 14-February 8, 2019	*Dampierre NPP Paris*	*Atucha NPP Maintenance Technical Support Assistant.*	*Peer Review (MA)*
February 19-21, 2019	*ENERGIE Electrabel Belgium*	*NA-SA Corporate Independent Oversight Head of Department*	*Corporate Peer Review action plant effectiveness review.*
March 4-7, 2019	*WANO Paris Centre France*	*NA-SA´s WANO Interface Officer*	*64th WANO Interface Officer Meeting*
March 4-8, 2019	*Forsmark NPP Sweden*	*Atucha NPP Shift Supervisor*	*Member Support Mission on Operational Risk Assessment*
March 4-22, 2019	*Krsko NPP Slovenia*	*NA-SA Corporate Engineering Management Mechanical Head of Division*	*Peer Review (EN)*
March 11-12, 2019	*Atucha NPP*		*Peer Review Follow Up Support Visit*
March 14-15, 2019	*Embalse NPP*		*Performance Improvement Action Plan Restart Review Support Visit*
March 19-20, 2019	*WANO Tokyo Centre Japan (Teleconference)*	*NA-SA Corporate Institutional Relations External Communication Head of Division*	*Communications Expert Group Meeting*

March 19-21, 2019	*Tricastin NPP Spain*	*Embalse NPP Engineering Process Assistant*	*Workshop on Fire Protection*
March 20-22, 2019	*WANO Paris Centre France*	*NA-SA´s Governor*	*76th WANO Paris Centre Governing Board*
April 5, 2019	*NA-SA Headquarter*		*Mid-Year Meeting*
April 8-26, 2019	*Yang Jiang NPP China*	*NA-SA Corporate Independent Oversight Head of Department*	*Peer Review (OA)*
April 8-26, 2019	*Yang Jiang NPP China*	*Embalse NPP Electrical Maintenance Head of Department*	*Peer Review (MA)*
April 8-26, 2019	*Yang Jiang NPP China*	*Atucha NPP Nuclear Safety Head of Section*	*Peer Review (EN)*
May 13-17, 2019	*Pickering NPP Canada*	*Embalse NPP Structures and Components Evaluations Head of Division*	*Benchmarking Visit on Equipment Reliability*
May 13-17, 2019	*Pickering NPP Canada*	*NA-SA Corporate Engineering Management Engineering Head of Department in Atucha NPP*	*Benchmarking Visit on Equipment Reliability*
May 13-17, 2019	*Pickering NPP Canada*	*Atucha NPP Civil Structures Evaluation Head of Section*	*Benchmarking Visit on Equipment Reliability*
May 13-17, 2019	*Dungeness NPP UK*	*Atucha NPP Operating Experience Head of Section*	*Benchmarking Visit on Risk Management*
May 13-17, 2019	*Dungeness NPP UK*	*Embalse NPP Operating Experience Head of Section*	*Benchmarking Visit on Risk Management*
May 28-29, 2019	*WANO Paris Centre France*	*Embalse NPP Plant Manager*	*Site Vice Presidents & Plant Managers Meeting*
May 29-June 14, 2019	*Hunterston UK*	*NA-SA Corporate Engineering Management Process Head of Division*	*Peer Review (EN)*
June 10-12, 2019	*NA-SA Headquarter*		*Human Resources and Knowledge Management Support Visit*
June 10-14, 2019	*Atucha NPP*		*Peer Review Follow Up*

The following activities will be performed in NA-SA during the rest of 2019:

- *65th WANO Interface Meeting in Ringhlas NPP, Sweden.*
- *66th WANO Interface Officer Meeting in WANO Paris Centre, France.*
- *Biennial General Meeting and 78th WANO Paris Centre Governing Board in London, UK.*
- *Secondee WANO Paris Centre.*
- *Annual Meeting in NA-SA Headquarter.*
- *Kick-off Meeting Corporate Peer Review in NA-SA Headquarter.*
- *Member Support Missions:*
  - *Fundamentals of production activities in the field of engineering support in Novovoronezh NPP, Russia.*
- *Peer Reviews:*
  - *Operations Reviewer in Asco NPP, Spain.*
  - *Fire Protection Reviewer in Asco NPP, Spain.*
  - *Radioprotection Reviewer in Forsmark NPP, Sweden.*
- *Workshops and Seminars:*
  - *Operator Fundamentals Workshop in WANO Paris Centre, France.*
  - *Corporate Peer Review Seminar in WANO Paris Centre, France.*
  - *Leaders in the Field Coaching Seminar in Angra NPP, Brazil.*

3.19.9. RADIOACTIVE WASTE MANAGEMENT

The legal framework applicable to radioactive waste is set up in the provisions of the National Constitution and the legislation adopted by the National Congress by National Law No. 24,804, that regulates the nuclear activity and other activities, and Act No. 25,018, that lays down the Radioactive Waste Management Regime.

In addition, Argentina has developed a legal and regulatory structure which complies with the safety provisions established in the Joint Convention. ARN is the Regulatory Body and CNEA is the Operating Organization for the final management of spent fuel and radioactive waste. Provisions have been adopted for NPP´s waste and spent fuel management (interim storage facilities) till a decision on their final management is taken.

3.19.9.1. RADIOACTIVE WASTE MANAGEMENT POLICY

The *Sixth* National Report to the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management (JCSSFMSRW) (2017) presents the following radioactive waste management policy:

- The radioactive wastes originated from all nuclear applications performed in the country, including wastes arising from the decommissioning of related facilities, will be managed safely, *guaranteeing the protection and rights of present and future generations as well as of the environment.*
- *The responsibility from radioactive waste management is born by the State through the Argentine Atomic Energy Commission where the generator will be responsible from the conditioning and safe storage of waste generated by the facility that operates until it is transferred to CNEA.*
- *The PEGRR that will be authorized periodically reviewed and audited by the National Congress.*
- *The sustainable procedure* to obtain and to manage the necessary financial resources in order to comply with the obligations arising from the performance of the assigned responsibilities with reference to this matter, considering that many of them imply costs deferred in time.

- A system for registry and preservation of information will be implemented, to ensure total tracking of inventories of radioactive waste generated and to be generated from all nuclear activities in the country.
- A public communication and information program will be implemented.

In agreement with this policy, the following additional factors have been taken into account:

- *The main responsibility for radioactive waste management lies in the National State through the Argentine Atomic Energy Commission (CNEA).*
- *The regulation and supervision of radioactive waste management are duties inherent to the National State performed by the Nuclear Regulatory Authority (ARN).*
- *The implementation of the policy on this matter will follow the guidelines of the National Radioactive Waste Management Program, with the responsibilities specified in Law No. 25,018, handling the radioactive waste management in the Republic of Argentina with an integrated perspective.*

In order to achieve its objectives, this National Radioactive Waste Management Program shall ensure the following:

- *Identification and assessment of accumulated and projected waste inventories.*
- *Adoption of the appropriate technological solutions for the safe management of such waste, with scientific-technological support.*
- *Definition of responsibilities and specification of obligations, and interrelations of the involved parties, from the generation of waste to the final stage of management.*
- *Definition of the required facilities for final disposal.*
- *Communication of its activities to the public and provision of the required information.*
- *Assessment of the costs associated to all these activities, determination of the financial sources and the financial and management methods.*

The establishment of the PEGRR implies the definition of the treatment methodology and the final disposal technological systems for the different types of waste. The review every three years of the Strategic Plan is conducted as set forth in the provisions of the Law and provides the opportunity to introduce the modifications originated by management optimisation in its technological aspects derived from scientific breakthroughs, or from the development of innovative technologies and eventual changes in the strategic definitions relative to spent fuel treatment.

The communication and information program intended for the public will provide the required information so that the population may value the scope of the proposed plans as well as their benefits, providing the adequate environment for public participation in subjects of their concern.

3.19.9.2. SPENT FUEL MANAGEMENT POLICY

The following paragraphs excerpted from the *Sixth* National Report to the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management *(2017)* present the Spent Fuel Management Policy:

"In Argentina, spent fuel is not considered radioactive waste. The Government exercises state ownership of special radioactive fission material contained in spent fuel from any origin: nuclear power plants and experimental, research and/or production reactors. (Section 2, Law No. 24,804). In this sense, the decision whether to reuse or not fissile material contained in spent fuel has to be adopted before 2030. At such time, the installation of the underground laboratory must have been started, which allows a deep geological repository to be designed and constructed. Said repository must be operative by 2060 (Strategic Plan – Law No. 25,018)".

With regard to spent fuel generated in the operation of research reactors or radioisotope production reactors and for which neither recovery nor further uses are envisaged, the strategy considers two alternatives:

- *Shipping to the country where the nuclear material was originally enriched, if possible.*
- *Wet interim storage. Afterwards, treatment and conditioning for final disposal.*

Here we may underline that due to the adhesion of Argentina to the RERTR Program (Reduced Enrichment for Research and Test Reactors) in December 2000, July 2006 and November 2007, all spent fuel from research and production reactors containing Highly Enriched Uranium (HEU) were exported to the Department of Energy of the USA (USDOE) in the frame of the Spent Nuclear Fuels from Foreign Research Reactors Acceptance Program".

3.19.9.3. SPENT FUEL TREATMENT AND STORAGE AT NPPs

3.19.9.3.1. CNA I NPP

Since the beginning of operation, spent fuel elements were stored in wet storage facilities. In order to optimize spent fuel pool storage capacity, NA-SA initiated in 2002 a compact storage project, which allowed 1,360 new positions.

CNEA and CNA I's Special Projects Division developed the project conceptual engineering for the Spent Fuel Elements Dry Storage (ASECQ).This project foresees SF transfer with a major decay deposited in the Pool Building I to an annex which will be the Transitory Dry Storage Building. This building will include vertical subterranean silos (subterranean silos in an upright position) and will be an extension of the *radiological* controlled area that will have the same features of the existing pool zone. It is expected not only to reach end of *the Phase A of the Long Term Operation*, but also to extend life operation of the plant for more than *3.5 years* of full power, enough time to implement a new Dry Storage System compatible with both plants (CNA I and II). According to the ASECQ project conceptual engineering, fuel will be located in a rectangular stainless steel storage unit (basket) with a capacity for nine SF; this unit will be hanging from a supporting grid in the upper part. In order to handle the baskets with SF, there will be a device (shield for transportation and lifting) to store the baskets and provide an appropriate shielding protection level to workers during transportation. Each silo will be made up of stainless steel, with a storage capacity of *one* baskets with 9 SF elements inside each. Some silos will include instrumentation for obtaining information about SF cladding temperature.

This facility is currently under construction *and it is expected to be operative in 2020*.

Regarding Argentinian Standards, available positions for a complete core should be kept in the pools during operation of the plant. With this requirement, the available positions in the pools were completed in October 2015.

Because it was foreseen that the project would not be completed in October 2015, when the available positions in the pools were completed, and in order not to affect the operation of CNA I, NA-SA considered the following alternatives:

a) Reorganization of the reactor's internal components placed in the decay pools hangers within the Pool Building I and II.

b) Transfer spent fuel elements from the Unit I to the Unit II.

The second alternative was chosen. The Conceptual Engineering, Detail Engineering and Preliminary Safety Report were developed. *In December 2017 the transfer of spent fuel elements was started. Nowadays, 1,435 spent fuel elements were transferred from CNA I to CNA II.*

3.19.9.3.2. CNE NPP

As it was informed in the report for the last Joint Convention (JCSSFMSRW) the CNE's spent fuel dry storage system (ASECQ) had till 2015 248 silos. The stored inventory at the end of *2018* was *118,520* fuel elements in *238* silos.

The spent fuel stored in the ASECQ silos has been included, at ARN's request, in the "Ageing Management Program for Components and Systems of the Nuclear Power Plant Associated to Nuclear Safety". The surveillance plan of canisters, internal cladding and concrete structure of all the ASECQ system silos was incorporated in the framework of this program. The surveillance, which commenced from its inauguration, continues to date and no abnormality has been observed in the behaviour analysis of these components. In addition to this surveillance action, a periodic measurement of aerosol and noble gases content inside the silos is conducted.

3.19.9.3.3. CNA II NPP

After its transport through the fuel transfer channel from the reactor building, spent fuels are placed in a vertical position in a spent fuel storage pool. Spent fuels hang from suspension beams, are stored in cooled demineralized light water until the stored policy in the long term is defined.

The spent fuel storage pools are reinforced concrete structures with stainless steel jacket. The design is such that no damages can occur to the concrete at a water temperature of 60ºC.

Three spent fuel storage pools have a 1,512 spent fuels *capacity and the other one has a 1,484 spent fuels capacity because it is using one hanger from the Unit I.* The decay heat of the spent fuels is transferred to the pool´s water and then is removed through a cooling system. In the spent fuel storage building, there is a room to place a transport vessel for spent fuels which isused to transport it outside the site. One of the spent fuel storage pools has capacity to store 733 irradiated fuel assemblies, to keep free space to accommodate the complete reactor core in case a off-loading is necessary.

Sub criticality is guaranteed due to the fact that spent fuels withdrawals from the reactor core are stored in safe geometric configurations which are stable, even after postulated accidents.

Nowadays, there are 1,435 spent fuel elements from the operation of the Unit I in the pool No. 3 of the UFA Building of the Unit II.

3.19.9.4. RADIOACTIVE WASTE PRACTICES AT NPPs

Radioactive waste management at NPPs was described in detail in the *Sixth* National Report presented to the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management. The following is a brief summary:

In the case of low level liquid wastes generated from NPPs, the management is different at each plant on account of the different technologies used.

- Liquid radioactive wastes generated at CNA I during operation and maintenance activities are collected in tanks, characterized and concentrated by evaporation; concentrates as well as sludge from the clean-up of tanks are immobilized in cement matrixes and conditioned in 200 litre drums.
- In the last eight years, it was not necessary to recondition the liquid radioactive waste and sludge in the storage tanks system (called TT11 and TT12) of the Plant due to the low generation rate of this type of waste and the wide storage capacity of the systems aforementioned.
- At CNE, liquid radioactive wastes originated from operation and maintenance activities are treated in resin beds, discharging the low activity current into the environment on the basis of planned and controlled procedures, following pre-established procedures and within the frame of authorized constraints of discharges.

Solid low level radioactive wastes at both NPPs are classified as compactable, non-compactable and structural. Compactable solid waste are collected in plastic bags and further compacted in 200 litre drums, following the corresponding CNEA guidelines. Non-compactable and structural waste are disassembled and sectioned prior to the conditioning in containers, if it is considered necessary, on a non–routine basis. *Sometimes, when the condition of the structural material justifies it, the element is decontaminated to minimize the generation of radioactive waste.* Such conditioned containers are stored temporarily at the facility.

Intermediate level radioactive solid wastes originated in the operation and maintenance activities of both NPP´s, consist mainly of filters and spent ionic exchange resins. These intermediate level radioactive solid wastes are stored at the facilities of each NPP. All the storage facilities are located at the NPP sites for an interim storage.

At the Ezeiza Waste Management Area (AGE), there is an especially designed interim storage facility where non-conditioned wastes may be stored prior to their processing as well as conditioned waste packages awaiting their transport and/or final disposal.

Since 2008, following a regulatory requirement, the operator developed a program for the characterization of solid and liquid radioactive wastes generated at the NPP and begun its implementation. This has as an objective, among others, the building of a data base for the design of future repositories. The program includes training of personnel and developing of a capability for waste handling and characterization. The program involves all wastes produced since the beginning of operation. This characterization also includes solid non compactable wastes as well as contaminated structural elements.

3.19.9.4.1. CNE life extension: waste management program

The waste management program developed for the activities to be performed during the CNE refurbishment is based on the following considerations:

- Regarding the waste generation, it is not expected that gaseous waste emissions exceed neither normal emissions, nor the liquid waste those of scheduled outages. The main radioactive waste corresponds to solid waste.
- Regarding the waste characterization, it has been estimated the radioactive inventory using samples, estimations, direct and indirect calculations in each case, estimated for a term of 50 years.
- Only 15% of the total volume of waste removed during plant refurbishing outage corresponds to activated waste, while the large percentage corresponds to contaminated waste.
- All deposits containing fuels channels (re-tubing) and SG´s tube bundles that will be replaced were designed and already built so that their integrity will be maintained for, at least, 50 years.

Regarding the re-tubing waste management, the CANDU Energy Inc. recommendations will be followed in the stages of treatment, conditioning and transfer. Such waste has no decontamination process, because it will be stored in vaults or containers once they are removed from the calandria.

For the waste coming from the SGs and moderator heat exchangers replacements, a pre-assembled concrete building with capacity for up to four SGs and two heat exchangers was built. The SGs´ pieces there installed will be, in themselves, a sealed source with a very low risk of contamination spread.

The rest of the waste, compactable, non-compactable and structural will be stored in the same place where now compactable wastes generated in the plant are stored, also new facilities for that type of waste were already constructed.

From January 2016 to December 2018, 1,692.2 m^3 of Radioactive Waste were generated (below the estimate).

3.19.9.5. RESEARCH AND DEVELOPMENT ACTIVITIES

Some activities have been started in the past, and must be continued in the coming years in order to achieve the expected results. Besides, others activities are analysed so as to create new projects or to be included in existing projects and, the necessary costs, time and human resources are estimated, taking into account the appropriate capacities of CNEA and other organizations.

As it was informed in the last Joint Convention (JCSSFMSRW) report, the ongoing activities of research and development are listed as follows:

- Corrosion studies for high level radioactive waste containers.
- Hydrology modeling in sedimentary environments and from the unsaturated zone.
- Hydrogeochemistry characterization studies: pedological, hydrogeological, groundwater and geomorphological in sedimentary environments, whoseknowledge shall be applied in determining the environmental baseline of new possible sites.
- Hydrological and hydrogeological characterization studies in fractured rocky environments of the frontal mountain range in Mendoza.
- Radiochemistry techniques selection for radioactive waste characterization.
- Development of equipment to verify conditioned waste quality by means of nondestructive tests, “Tomographic Gamma Scanner (TGS)”.
- Draft evaluation studies of the following spent ion exchange resin processing alternatives:
- thermal and biotic degradation methods.
- Studies about conditioning by cementation of liquid radioactive waste stored in the AGE.

- Review of the features of SF generated from spent research reactors and evaluation for their future management.
- Durability study of cement-based materials as an engineering barrier to build the low level radioactive waste repository.
- Studies about contaminated hydraulic oil solidification procedures.
- Feasibility study about monitoring through CNE's spent fuel dry silos by tomographic images.
- R&D activities aimed at consolidating the design of a Research Reactor Spent Fuel.
- Transport Package (called RLA4018), certified by the ARN.
- Monitoring activities for the corrosion of aluminum-based irradiated nuclear fuel under water.

3.19.9.6. MINIMIZATION OF RADIOACTIVE WASTES

The policy of the NPP´s Licensee is to optimize the impact on the workers, public and the environment as a result of its operation. Therefore, one of its main goals is to keep the radioactive waste generation to the minimum practicable, and thus, an efficient and effective ALARA program has been implemented for both NPPs. These practices include:

- Detailed planning of the activities that involve a significant waste generation and/or individual doses.
- Mock-up training for the above mentioned activities.
- Design of specific tools and shielding to be used to handle wastes in high radiation fields.
- Compliance with segregation procedures of radioactive wastes.
- Damaged fuel elements are immediately withdrawn from the core and isolated.
- Personnel training in the application of radioactive waste management procedures.
- Recycling and reuse of contaminated or active materials.
- Measurement, characterization, segregation and compaction of radioactive wastes.

In addition, the Licensee has taken important actions to reduce the radioactive waste generated.

Presently, in CNA I, slightly-enriched uranium fuel elements (0.85%) are being used and consequently, the generation of spent fuel elements has been reduced. Further, changes of the core channels bearing "stellite" reduced the ^{60}Co generation and the activity in operational wastes.

Further details were presented in the previous National Reports to the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management (2003, 2005, 2008, 2011, 2014 and 2017). Any further information can be obtained from the above mentioned National Reports that can be downloaded from: http://www.cnea.gov.ar/PNGRR-Convencion-Conjunta-Seguridad.

3.19.10. COMPLIANCE WITH THE OBLIGATIONS IMPOSED BY THE CONVENTION

The information contained in this and other Articles demonstrates that Argentina complies with the obligations imposed in Article 19 of the Convention on Nuclear Safety.

CHAPTER 4

PLANNED ACTIVITIES TO IMPROVE SAFETY

The ARN and NA-SA identified a number of challenges during this reported period. Those challenges lead to actions with regards to both safety-related issues and regulatory issues in order to maintain and improve the safety level of the Argentine nuclear power plants during their remaining operating lives, as well as to improve the regulatory framework. In the following paragraphs, the intended measures are presented in a summary form.

4.1. CNA I LONG TERM OPERATION PROGRAMME

As it was mentioned in Section 3.6.5.2. and in the previous National Safety Report, NA-SA has decided to undertake a Long Term Operation of CNA I.

The activities to be performed for demonstrating a safe continued operation (Implementation Plan) are being identified through a methodological approach established by the IAEA SSG-48, "Ageing Management and Development of a Programme for Long Term Operation of a Nuclear Power Plant".

The regulatory expectations and requirements were already issued by ARN and are being considered in the development of the Implementation Plan.

Engineering studies are currently being developed and implementation of design upgrades are being analysed based on the PSR's Global Assessment methodology, according to IAEA SSG-25. Some examples are:

- *Analysis of Design and Construction Standards of CNA I and comparison with current KTA standards.*
- *Consequential Failure Analysis.*
- *Revalidation of the identified TLAA.*
- *Leak Before Break analysis in main pipes.*
- *Stress & Fatigue analysis in main pipes, FIAT valve pressure casing and moderator pump pressure casing.*
- *Condition assessments of structures and components belonging to safety class 3 and 4 systems.*

The Global Assessment and the derived Implementation Plan are expected to be finished and submitted to ARN by March 2020.

In order to ARN review & assess the Implementation Plan, a methodology based on the adequacy of the plant's defence in depth by reference to all levels as defined in IAEA SSR2/1 Rev.1 is being developed. The regulatory decision to proceed with the LTO – Phase B of CNA I will be based on the acceptance of the Implementation Plan at the extent it can improve the robustness of the current Defence in Depth Concept enhancing plant safety, as far as reasonably practicable.

ARN decision making is reliant on the demonstration that the remaining overall plant risk, as derived from the proposed Implementation Plan, is acceptable.

4.2. NORMATIVE FRAMEWORK REVIEW

The Regulatory Body agreed with the Vienna Declaration on Nuclear Safety and its adoption in order to prevent accidents with radiological consequences and to mitigate such consequences should they occur. In this sense, as was mentioned in Sections 1.4.2.3. and 3.7.2., ARN is carrying out a normative framework review that includes addressing the Vienna Declaration in national standards.

4.3. PREPARATION FOR HOSTING AN IAEA IRRS MISSION

As was mentioned in Section 2.11. of this National Safety Report, several activities are being performed by the Regulatory Body in preparation for receiving the IAEA IRRS (Integrated Regulatory Review Service) mission.

The IRRS mission is scheduled for May 4th, 2020, as it can be seen at IAEA's webpage.

Completion of the self-assessment is considered by ARN´s staff a valuable task contributing to analyze the organization and its practices in a systematic way, allowing the identification of actions for continuing improvement.

4.4. LICENSING OF ATUCHA III NPP: HPR-1000

In section 1.3. of this National Safety Report, it was mentioned that a Memorandum of Understanding was signed between ARN and NA-SA oriented to the construction of a fourth NPP.

The fourth plant will be a HPR PWR unit (or Hualong I-1000), with Fuqing unit 5 under construction in China taken as a reference design, and including design changes according relevant updates of argentine and IAEA Safety Standards. Thus, the licensing and project development of this new NPP has to be considered as a first-in-a-country type.

During the licensing process, design documentation as well as manufacturing documentation needs to be efficiently and effectively reviewed. In order to do so, ARN is planning the following activities:

- *To promote greater cooperation of regulators involved in licensing activities of same NPP technology.*
- *To promote harmonization of safety requirements and standardization of reactor design.*
- *To participate in groups or programs oriented to share licensing experiences and results obtained in another country.*

4.5. INDEPENDENT NUCLEAR OVERSIGHT

In October 2017, the Licensee of CNA I, CNA II and CNE implemented a Corporate Independent Oversight process based on WANO / IAEA guidelines.

The objective of the Corporate Independent Oversight is to promote excellence in the operation of nuclear power plants throughout the company and to provide the Chief Nuclear Officer, corporate managers and the board of directors with a permanent perspective of the performance of nuclear power plants and corporate organization compared to the industry, focusing mainly on nuclear safety, plant reliability and emergency preparedness.

A dedicated organization has been set up with roles and responsibilities at plant and corporate level. The process includes daily activities at plant level, planed reviews based on a predefined annual schedule and an escalation process of plant issues. Review results (daily and planned reviews) are presented at the appropriate management level according to the significance of the identified issues. Corrective actions are defined to address the identified areas for improvement, which are tracked to completion by Corporate Independent Oversight.

4.6. ATUCHA SPENT FUEL DRY STORAGE

Base on the spent fuel pool capacity and planned operation of CNA II, NA-SA has decided the construction of a Spent Fuel Dry Storage. The technical specifications and bidding conditions have been recently finished. The storage will be ready by the end of 2023 and the main characteristics will be:

- *Independent installation, not requiring services from the nuclear power plant.*
- *Modular storage.*
- *Capacity of about 16,000 fuel elements from both Units of Atucha NPP*
- *80 years of fuel confinement.*
- *Natural cooling for the modules.*

ANNEXES

ANNEX I

CONCLUSIONS ABOUT ARGENTINA DURING THE SEVENTH REVIEW MEETING ON THE CONVENTION ON NUCLEAR SAFETY

RAPORTEUR REPORT

I.1. EXECUTIVE SUMMARY

Argentina has three nuclear power reactor units in operation. Two of these, Atucha I (CNA I) and Atucha II (CNA II), are pressurized heavy water reactors (PHWR) and one, Embalse (CNE), is PHWR of Canada Deuterium Uranium (CANDU) type. A fourth nuclear power reactor unit is planned.

A low power prototype reactor, Central Argentina de Elementos Modulares (CAREM), is under construction at Atucha site.

Seven out of nine Challenges from the 6th Review Meeting have been closed.

The Country Group highlights the following measures to improve safety in Argentina´s national nuclear programme:

- Several changes to the regulatory framework and the national nuclear programme have been made.
- Installation of a new physical protection system (PPS) at Atucha site. At Embalse, improvement of the physical protection system is ongoing. A substantial proportion of the work at Embalse has been completed.
- Identifying / implementing alternative water supply systems to face severe accident situations caused by the loss of heat sink at all nuclear power plants in operation.

The Country Group highlights the following regarding international peer review missions of Argentina:

- Argentina has been relying mostly on World Association of Nuclear Operators (WANO) peer reviews and has not invited any Operational Safety Review Team (OSART) missions.
- A Safe Long Term Operation (SALTO) mission will be conducted at CNA I during Q3 2017.
- Argentina has invited Integrated Regulatory Review Service (IRRS) mission, planned during Q4 2018.

The Country Group identified the following challenges for Argentina:

- **Challenge 1:** The Regulatory Authority to prepare and host the IRRS Mission in 2018 (**new**).
- **Challenge 2**: The country to prepare and conduct relevant activities related to Atucha I Life Extension with respect to the SALTO mission (**new**).
- **Challenge 3**: Resolution of issues with Atucha I and II RPV in-vessel retention and external cooling arising from FORO stress tests (**new**).
- **Challenge 4**: The Regulatory Authority to conduct licensing activities on CAREM 25 small modular prototype reactor under construction following Principle 1 of the VDNS.
- **Challenge 5:** External emergency control center located far from Embalse NPP.

In addition, the Country Group identified seven Areas of Good Performance.

The Country Group concluded that Argentina:

- Submitted a National Report, and therefore complies with Article 5 and in time following Rule 39 of INFCIRC/573 Rev. 6.
- Attended the 7th CNS Review Meeting, and therefore complies with Article 24.1.
- Held a national presentation and answered questions, and therefore complies with Article 20.3.

I.2. BASIC INFORMATION ON ARGENTINA'S NUCLEAR PROGRAMME

Argentina has three nuclear power reactor units in operation. Two of these, Atucha I (CNA I) and Atucha II (CNA II), are pressurized heavy water reactors (PHWR) and one, Embalse (CNE), is PHWR of CANDU type.

A fourth power reactor unit is planned according to a Memorandum of Understanding that was signed between Argentina and China in 2015. The 4th plant would be a PHWR CANDU unit, with Qinshan Phase III units taken as a reference design, with changes according relevant updates of Argentine and IAEA Safety Standards, and post Fukushima safety improvements.

Activities related with the CNE´s Life Extension Project, to extend the plant life for another 25 years of full power operation as well as to increase the electrical power in about 35 MW, are presently being carried out. Different design changes were introduced to improve safety, including post-Fukushima requirements for severe accidents. The refurbishment shutdown started on December 31st, 2015.

A low power reactor prototype, Central Argentina de Elementos Modulares (CAREM), is under construction at Atucha site.

I.3. FOLLOW-UP FROM PREVIOUS CNS REVIEW MEETING

Seven out of nine challenges from the 6th Review Meeting have been closed. The open ones are the following:

Challenge 4: CAREM-25 prototype licensing

Argentina reported that ARN in December 2014 issued the Authorization for the Construction of Nuclear Module of CAREM 25.

During 2015 ARN continued with licensing tasks, related to design information of Structures, Systems and Components Relevant to Safety (SSCRS), where the reactor pressure vessel (RPV) components, Safety Systems, Protection System and Contention are included among others. In addition, in this period ARN lead an inspection regime at the site of the reactor CAREM 25, related to the construction of civil structures (currently underway).

For the next period (2016 - 2019), ARN expects to go forward with the licensing process, according to the development of CAREM 25 project. Audits and inspections will be focused on SSCRS of regulatory interest, which were defined in the "Authorization for Use of Site and Construction" (AUSC).

According to CNEA project schedule, the application to achieve the next step of the licensing process, "Authorization of Fuel Load", will be submitted in 2018.

Challenge 7: External emergency control center located far from Embalse NPP

Argentina reported that the design of the CNE External Emergency Control Centre, called Municipal Emergency Operational Centre (MEOC) is in progress.

The MEOC building will be 10 km outside the radius of Embalse NPP by ARN requirement. The construction site will have a quick access and more than one way of alternative access.

I.4. MEASURES TO IMPROVE SAFETY

I.4.1. Changes to the regulatory framework and the national nuclear programme

Since the last Review Meeting, the Country Group took note of several changes to the regulatory framework and the national nuclear programme. Some examples are:

- Resolution ARN No. 352/13, authorization for Use of the Site and Construction of CAREM Prototype Reactor.
- Resolution ARN No. 238/14, issued the Commissioning License to the Nuclear Power Plant Atucha II (CNA II).
- Resolution ARN No. 302/16 issued the Operating License to the Nuclear Power Plant Atucha II (CNA II).
- The ARN has continued with the review of its standards related to nuclear power plants.
    - The Regulatory Standard AR 10.10.1. “Site Evaluation for Nuclear Power Plants” has recently been developed and put into force, taking into account the lessons learned from the Fukushima accident and the corresponding IAEA standards.
    - The Regulatory Guide AR 13 “Storage of Radioactive Waste” to be applied to type I installations was developed and put into force in order to facilitate the Regulatory Standard fulfilment.
- There are other guides under development, e.g. for the implementation of the Regulatory Standard AR 3.1.3. "Radiological Criteria Relating to Accidents in Nuclear Power Plants" and for the safety classification of Systems, Structures and Components (SSC).

I.4.2. Safety improvements for existing nuclear power plants

The Country Group took note of several implemented and planned safety measures for existing nuclear power plants in Argentina.

Argentina reported on several safety improvements, which have been implemented at CNA I, CNA II and CNE. The improvements include, but are not limited to:

- A new Physical Protection System (PPS) for Atucha site, including both CNA I and CNA II, was installed. A substantial portion of improvement work on CNE physical protection system is already completed, including the complete overhauling of the secondary entrance.

- At CNA I an alternative system was implemented to supply water to the spent fuel storage pools; it consists of a well water pumping system. In addition, the firefighting system of the construction site of CNA II was identified as an alternative water reservoir. At CNE, additional alternative water sources to face severe accident situations caused by the loss of heat sinks were implemented.

I.4.3. Response to international peer review missions

According to Argentina´s National Report, during the period 2013-2016, following WANO missions have been conducted in the Argentinian NPPs in operation and in commissioning: A WANO Peer Review for CNE was conducted in September 2014, a WANO Pre-Start-Up Peer Reviews for CNA II was conducted in September 2013 and a Corporate Peer Reviews was conducted in November 2014.

During 2016 a WANO Peer Review Follow-up mission has been conducted at CNE. NA-SA has provided experts on two WANO Technical Support missions, benchmarking visits, at Angra in Brazil, on modification management, and at Vattenfall in Sweden, on independent nuclear oversight.

For CNA I, a Safe Long Term Operation (SALTO) peer review has been requested. The preparatory meeting took place in March 2016, and a Pre-SALTO mission was conducted in September 2016. The following steps have been scheduled:

- Preparatory meeting 2 – Q1 2017,
- SALTO mission – Q3 2017, and
- Follow-up SALTO mission – Q3 2020.

No Operational Safety Review Team (OSART) missions have been planned nor requested. According to Argentina´s National Report, the reason is that the CNE life extension project has required a great amount of preparatory work by specialists in different areas, and they will continue requiring an extraordinary effort in the next few years.

In December 2014, the head of the ARN informed the IAEA the decision to initiate a process to receive an Integrated Regulatory Review Service (IRRS) mission in the future, convinced that this step will be a relevant contribution to the demonstration, at national level, of the implementation of the objectives of the Safety Action Plan signed by all IAEA´s Member States. The mission is planned to be conducted during Q4 2018.

Argentina did not specify, in its National Report, any implemented or planned measures in response to international peer reviews.

The Country Group took note of the following planned international peer review mission:

- Argentina has invited an IRRS mission, which is planned during Q4 2018.
- Argentina has been relying mostly on WANO peer reviews and has not invited IAEA OSART missions to their nuclear power plants.

I.5. IMPLEMENTATION OF THE VIENNA DECLARATION ON NUCLEAR SAFETY (VDNS)

On February 9th 2015, the Contracting Parties adopted INFCIRC 872, “Vienna Declaration on Nuclear Safety”, which is a commitment to certain principles to guide them in the implementation of the CNS' objective to prevent accidents and mitigate their radiological consequences, should they occur. The Contracting Parties agreed to discuss the principles of the Vienna Declaration on Nuclear Safety in their National Reports and in the subsequent Review Meetings.

The Argentinian National Report addressed all three principles of VDNS.

I.5.1. Implementation of the VDNS’s principle on new nuclear power plants

The first principle of the VDNS is:

“New nuclear power plants are to be designed, sited, and constructed, consistent with the objective of preventing accidents in the commissioning and operation and, should an accident occur, mitigating possible releases of radionuclides causing long-term off site contamination and avoiding early radioactive releases or radioactive releases large enough to require long-term protective measures and actions.”

Argentina´s definition of a new nuclear power plant, as for the purpose of the Vienna Declaration, is any nuclear power plant that has initiated the licensing process after 2010. Nowadays, it includes CAREM 25 and the country´s 4th Nuclear Power Plant.

Argentina reports, that its national requirements and regulation incorporate appropriate technical criteria and standards.

The ARN is currently engaged in the licensing process of the CAREM 25 Reactor.

CAREM 25 design features have an enhanced implementation of the Defence in Depth (DiD) concept, and can therefore, according to the ARN, be considered to be an example of how the basic objective in the Vienna Declaration could be implemented in future projects. A summary of basic design aspects of CAREM 25 Reactor in relation with DiD concept is presented in Argentina´s national report.

The Country Group made the following observations:

- Argentina´s definition of a new nuclear power plant, as for the purpose of the Vienna Declaration, is any nuclear power plant that has initiated the licensing process after 2010. Nowadays, it includes CAREM 25.
- The Argentinian National Report and Presentation provided information on how the country meets the safety objective of the VDNS.

I.5.2. Implementation of the VDNS's principle on existing nuclear power plants

The second principle of the VDNS is:

"Comprehensive and systematic safety assessments are to be carried out periodically and regularly for existing installations throughout their lifetime in order to identify safety improvements that are oriented to meet the above objective. Reasonably practicable or achievable safety improvements are to be implemented in a timely manner."

The ARN has, since 2003 required a Periodic Safety Review (PSR) as a condition for license renewal. The requirement to develop PSR was included in the operating license of CNA I, CNA II and CNE. It was established that PSRs have to be developed every 10 years, with the scope described in the IAEA Safety Standard Series No. NS-G-2.10 "Periodic Safety Review of Nuclear Power Plants". Approval of PSR results by ARN is a necessary condition for license renewal.

CNA I presented its first PSR in 2014.

In the case of the CNE, due to the fact that the plant is engaged in a plant life extension program, PSR is being developed as part of the safety assessments for that project. However, aging evaluations were completed and design improvements to be introduced during refurbishment outage emerged from the results of safety evaluation made in other CANDU plants. As a condition for renewal of the operating license for the second cycle, ARN requested a PSR with a comprehensive scope.

Argentina reports, that its national requirements and regulation address the application of the principles and safety objectives of the Vienna Declaration to existing nuclear power plants and require the performance of periodic comprehensive and systematic safety assessments of existing plants. Also, reasonably practicable / achievable safety improvements are to be implemented in a timely manner.

The Country Group made the following observations:

- The Argentinian regulatory framework requires Periodic Safety Reviews to be conducted.
- The Argentinian regulatory framework requires safety improvements to be implemented in a timely manner on existing nuclear power plants.

I.5.3. Taking into account IAEA Safety Standards and other international Good Practices in the national requirements and regulations addressing the VDNS principles

As Argentina has reported in its previous National Reports, the ARN performed a process of harmonization between the Argentinean Regulatory Standards and the IAEA Safety Standards. As result of such harmonization, it was concluded that Argentine Regulatory Standards are consistent with IAEA´s corresponding standards, taking into account that the ARN has mainly adopted a performance criterion.

Argentina participates actively in the IAEA standards committee´s activities and particularly in the international efforts to take account of the lessons learned from the Fukushima accident, in Nuclear Safety Convention, as well as to maximize the benefit of the mentioned lessons learned.

Argentina reports that the ARN agreed with the Vienna Declaration on Nuclear Safety and adopted it in order to prevent accidents with radiological consequences and to mitigate such consequences should they occur. In this sense, the ARN decided to carry out a normative framework review that includes addressing the Vienna Declaration in national standards.

The Country Group made the following observations:

- The Argentinian national legislation and the regulatory framework takes into account the relevant IAEA Safety Standards.

I.5.4. Issues faced by Argentina in the implementation of the VDNS

Argentina does not report facing any specific issues in applying the Vienna Declaration principles and safety objectives to its existing fleet or new builds of nuclear power plants. Implementation of the VDNS principles and goals into the national legislation is underway.

I.6. CHALLENGES

The Country Group identified the following challenges for Argentina:

- **Challenge 1:** The Regulatory Authority to prepare and host the IRRS Mission in 2018 (**new**).
- **Challenge 2**: The country to prepare and conduct relevant activities related to Atucha I Life Extension with respect to the SALTO mission (**new**).
- **Challenge 3**: Resolution of issues with Atucha I and II RPV in-vessel retention and external cooling arising from FORO stress tests (**new**).
- **Challenge 4**: The Regulatory Authority to conduct licensing activities on CAREM 25 small modular prototype reactor under construction following Principle 1 of the VDNS.
- **Challenge 5:** External emergency control center located far from Embalse NPP.

I.7. SUGGESTIONS

The Country Group identified no Suggestions for Argentina.

I.8. GOOD PRACTICES AND AREA OF GOOD PERFORMANCE

During the peer review of Argentina´s National Report, the Contracting Parties were invited to recommend Good Practices and to highlight Areas of Good Performance.

The Country Group identified no Good Practices.

The following Areas of Good Performance of Argentina were commended by the Country Group:

- **Area of Good Performance 1**: The use of PSA in the CNE´s plant life extension (PLEX) program.
- **Area of Good Performance 2:** Results of the FORO stress tests were conveyed to the heads of the governments.
- **Area of Good Performance 3:** Process of Internal Technical Support Missions (MISTI) implemented by the licensee.
- **Area of Good Performance 4:** IV NPP pre-licensing process by early definition of licensing basis documents defining requirements and expectations.
- **Area of Good Performance 5:** Licensee capabilities to perform analysis taking into account specific phenomenology in severe accident progress for Atucha plants.
- **Area of Good Performance 6:** Improvement in Emergency preparedness for both sites by the licensee considering lessons learned from Fukushima.
- **Area of Good Performance 7:** PRACS: Program of Safety Culture Strengthening has been implemented since 2013. The objective of the program is to create a bridge between the concepts of Safety Culture and the actual performance of the plants.

ANNEX II

ANSWER TO QUESTIONS OR COMMENTS

NATIONAL NUCLEAR SAFETY REPORT – 2016

No. 1
Country: Austria
CNS-REF.-ART.: General
PAGE OF REPORT: 15
CHAPTER OF NAT. REPORT: Section 2.8.

The process of Internal Technical Support Missions (MISTI), implemented by NA-SA, can be seen as area of good performance for Argentina. The process has been applied mainly between NA-SA plants, including support organizations. Since 2011, when the first MISTI was initiated, a number of 23 MISTI missions were performed?
Argentina appreciates the comment from Austria.

No. 2
Country: Canada
CNS-REF.-ART.: General
PAGE OF REPORT: 14
CHAPTER OF NAT. REPORT: Section 2.5.

The report states, “as a condition for licensing the CNE life extension, to carry out a periodic safety review…”. (CNE, Central Nuclear Embalse)… Can the Contracting Party clarify if the PSR is required to inform the regulatory decision on licence renewal, or as a condition following licence renewal?
In Argentina, license renewal is based on successful completion of PSR. Therefore, for CNE life extension, it is required to inform the regulatory decision.

No. 3
Country: Canada
CNS-REF.-ART.: General
PAGE OF REPORT: 6
CHAPTER OF NAT. REPORT: Section 3.6.3.2.2.

The report states that “An interconnection between CNA I /CNA II normal electric bars will be maintained manually activated. Among the improvements that are being analyzed after Fukushima, it is considered that CNA I´s EPS can be a backup for CNA II through this interconnection. These measures are still under study, and up to now no calculations on their impact on global core damage frequency have been done.” Can you elaborate on the progress and implementation of this interconnection?
The current status is the following:

- The electrical interconnection between CNA I and CNA II is now manual and it has been incorporated into the operational instructions following a blackout. This interconnection can be in both directions in order to supply power to the normal bars. For other purpose, such as for closing the switches, previous actions are needed, according to the load flow.
- A conceptual engineering assessment has been performed in order to supply power to a CNA II secured bar using diesel generators from CNA I. This action is foreseen in an Emergency Power Case of both plants, plus unavailability of the 4 diesel generators of CNA II and having the 3 CNA I diesel generators available.

No. 4
Country: Canada
CNS-REF.-ART.: General
PAGE OF REPORT: 6
CHAPTER OF NAT. REPORT: Annex IV 12

Can you please clarify for the design of the CAREM 25 what is meant by “Emphasis has been given since the design genesis to prevention of core degradation accidents by means of passive safety features, guarantying no need of active systems or operator actions for a period of at least 36 hours.” Are batteries considered part of the passive safety features? If so, are there any additional strategies being considered to extend the life of the batteries (i.e., load shedding)?
Batteries are considered part of the passive safety features. Moreover, the valves needed to control the main line of the passive safety features go to a safe position in case of loss of electrical power. This means that these passive safety features are triggered in case of loss of the batteries.

No. 5
Country: Finland
CNS-REF.-ART.: General
PAGE OF REPORT: 5
CHAPTER OF NAT. REPORT: Section 1.4.1.

The report describes the process for stress tests and its peer review within FORO. Could it be clarified if the stress test national report and its peer review results were made publicly available?

National Reports from Argentine, Mexican, Brazilian and Spanish NPP stress tests were published at FORO web page in Spanish: http://www.foroiberam.org/web/guest/publicaciones/evaluacion

The FORO presentation during the VI Meeting to Review Safety Convention, in English, is available at this web page too.

No. 6
Country: Finland
CNS-REF.-ART.: General
PAGE OF REPORT: 11
CHAPTER OF NAT. REPORT: Section 2.1.

The report indicates that several WANO peer review missions have been conducted in Argentina during the review period. However, no OSART missions have been conducted. Is there a plan to invite also OSART missions to Argentina?
In order to request an OSART mission to Argentina in the future, in 2016 two regional workshops on operational safety have been conducted in Argentina focused on Supply Chain (procurement) and Nuclear Safety Oversight. The following meetings are scheduled to take place during 2017 in Argentina:

- Knowledge management for CNEA and NASA
- Supply chain from engineering and maintenance aspects
- Participation of experts from Regulators, Utilities or TSOs in OSART missions or workshops as Observers

In addition, the participation of Argentina in three meetings to be conducted in Brazil and Mexico during 2017 is planned.

No. 7
Country: Finland
CNS-REF.-ART.: General
PAGE OF REPORT: 21
CHAPTER OF NAT. Section 2.15.

The report indicates that a process has been initiated to invite an IRRS mission to Argentina. When is the IRRS mission planned to take place?
The IRRS mission in Argentina is planned to take place in the 4th Quarter of 2018 (4Q2018), according to the IAEA Schedule for missions for 2018: https://gnssn.iaea.org/regnet/irrs/Pages/Events.aspx

No. 8
Country: Luxembourg
CNS-REF.-ART.: General
PAGE OF REPORT: VDNS
CHAPTER OF NAT. Sections 1.4.2. and 3.7.2.2.

Please elaborate on the following aspects related to the VDNS:
1) How do you define 'a new nuclear power plant'?
2) How does your national requirements and regulations incorporate appropriate technical criteria and standards to address the objective of preventing accidents in the commissioning and operation of new nuclear power plants?
3) How do your national requirements and regulations incorporate appropriate technical criteria and standards to address the objective of mitigating against possible releases of radionuclides causing long-term offsite contamination and avoiding early radioactive releases or radioactive releases large enough to require long-term protective measures and actions?
4) How do your national requirements and regulations address the application of the principles and safety objectives of the Vienna Declaration to existing NPPs?
5) Do your national requirements and regulatory framework require the performance of periodic comprehensive and systematic safety assessments of existing NPPs – if so, against what risk/engineering objective or limit are these judged and can you give practical examples?
6) How do your national requirements and regulations take into account the relevant IAEA Safety Standards throughout the life-time of a Nuclear Power Plant?
7) What issues have you faced or expect to face in applying the Vienna Declaration principles and objectives to your existing fleet or new build of Nuclear Power Plants?
The IRRS mission in Argentina is planned to take place in the 4th Quarter of 2018 (4Q2018), according to the IAEA Schedule for missions for 2018: https://gnssn.iaea.org/regnet/irrs/Pages/Events.aspx

1) How do you define 'a new nuclear power plant'?

ARN will consider as a New NPP to every nuclear power plant that has initiated the licensing process after 2010. This includes the CAREM reactor and the 4th NPP that is expected to initiate the licensing process in 2017.

2) How does your national requirements and regulations incorporate appropriate technical criteria and standards to address the objective of preventing accidents in the commissioning and operation of new nuclear power plants?

Argentine Regulatory Framework is not prescriptive.

The objectives stated in the set of standards are fully compatible with the safety concepts and objectives included in the Vienna Declaration. The compliance with technical criteria is verified through the consolidated practice of review of the licensing basis documents and regulatory requirements.

In particular, Argentine standards includes among others AR 3.1.3 "Criterion Curve" and AR 3.2.1 which deals with technical criteria to achieve an acceptable safety level necessary to enter into commissioning phase.

AR 3.1.3 "Criterion Curve", establishes a limit to the risk from any accident situation through a probabilistic quantification.

In AR 3.2.1 standard "Safety Criteria in the design of NPPs", the implementation and compliance with the Defence in Depth concept is explicitly required.

In order to maintain a safety level during operation, AR 3.9.1 “General safety criteria for the operation of NPPs” standard establishes a set of organizational, procedural, maintenance and surveillance requirements to be fulfilled by the utility during the whole plant life.

3) How do your national requirements and regulations incorporate appropriate technical criteria and standards to address the objective of mitigating against possible releases of radionuclides causing long-term offsite contamination and avoiding early radioactive releases or radioactive releases large enough to require long-term protective measures and actions?

For new NPPs, a full study of PSA of levels 1, 2 and 3, is required in order to fulfill AR 3.1.3 standard and then any findings that lead to possible releases are identified and appropriate measures are analyzed. Besides, severe accident management is developed.

4) How do your national requirements and regulations address the application of the principles and safety objectives of the Vienna Declaration to existing NPPs?

The Safety Review of existing Nuclear Power Plants in Argentina is an activity that has been behind the periodical renewal of operating licenses. These reviews were historically conducted in “case by case” terms and with the aim of improving, as much as achievable, the safety level of the plant. The terms of the Safety Review were revisited under the light of SSG-25 “Periodic Safety Review for Nuclear Power Plants” since its issuance a few years ago.

In the frame of IAEA’s action plan, within the Iberoamerican Forum of Radiological and Nuclear Regulatory Agencies (FORO), an application of Fukushima lessons learned was implemented. Argentina has carried out comprehensive stress tests aimed to determine the existing safety margins to cope with extreme events, analyzing their behavior and the consequences for design extension conditions scenarios, such as station black-out and the loss of ultimate heat sink for a long term, as well as the capacity to manage such accidents.

Argentina has also carried out special safety reviews and upgrades of the NPPs taking into account the operating experience.

Argentina does not have any specific regulation dealing with periodic safety review. As stated before, these reviews were conducted using case by case methodology, and in all cases the identified improvements have been enforced via requirements issued by ARN with time constraints.

5) Do your national requirements and regulatory framework require the performance of periodic comprehensive and systematic safety assessments of existing NPPs – if so, against what risk/engineering objective or limit are these judged and can you give practical examples?

Answer in the above question.

6) How do your national requirements and regulations take into account the relevant IAEA Safety Standards throughout the life-time of a Nuclear Power Plant?

Our national regulation is compatible and in line with the relevant IAEA safety standards. Any identified gap in our national regulation is foreseen to be covered taking into account the IAEA standards and recommendations.

7) What issues have you faced or expect to face in applying the Vienna Declaration principles and objectives to your existing fleet or new build of Nuclear Power Plants?

We have not faced any major issue in applying the Vienna Declaration principles and objectives.

No. 9
Country: Poland
CNS-REF.-ART.: General
PAGE OF REPORT: 12
CHAPTER OF NAT. Section 2.2

Why the physical protection system was not completed (lack of perimeter sensors, temporary measures in access control system) before commissioning of CNA II?

The original design of CNA II did not consider the Physical Protection System. It was added in the commissioning stage. Priority was given to the construction of the double perimeter fence and to the access control system. The rest is currently under implementation.

No. 10
Country: Portugal
CNS-REF.-ART.: General
PAGE OF REPORT: 31
CHAPTER OF NAT. Section3.6.3.2.

What percentage of your NPP's already have a containment venting-filtration system installed?
There are 3 reactors in operations in Argentina. Embalse NPP is currently in long shutdown for refurbishment and a filtered containment venting system is planned to be installed during the shutdown. The installation of this system for Atucha Unit I and II is under evaluation.

No. 11
Country: Portugal
CNS-REF.-ART.: General
PAGE OF REPORT: 31
CHAPTER OF NAT. Section3.6.3.2.

What percentage of your NPP's already have autocatalytic hydrogen recobiners installed in the containment?
All the NPP's, CNA I, CNA II and CNE have Auto-catalytic Recombiners. In the case of Embalse, it is being installed and will be ready for the extension life.

No. 12
Country: Slovakia
CNS-REF.-ART.: General
PAGE OF REPORT: 21
CHAPTER OF NAT. Section 2.15.

Argentina invited an IRRS mission in 2016. Please provide information on outcomes of the IRRS mission and on the planned measures to implement its findings during the national presentation.
The IRRS mission in Argentina is planned to take place in the 4th Quarter of 2018 (4Q2018), as it was mentioned in the National Report, it was requested in 2016. The outcomes of the mission and the planned measures to implement its findings will be presented at the 8th Review Meeting.

No. 13
Country: United Kingdom
CNS-REF.-ART.: General
PAGE OF REPORT: 8, 154, 158-159
CHAPTER OF NAT. Section 1.4.2.1, 3.18.3.1.1, 3.18.3.2.2 & 3.18.3.3.1

Section 1.4.2.1 of the National Report describes a 4-level scheme of Defence in Depth (DiD) arrangements. Could Argentina expand on the text in Section 3.18.2 and under 'design improvements implementation' to explain in more detail to what extent the existing NPPs meet those levels now?
Argentinean NPP's have met the requirements of the Defense in Depth levels, long before the Fukushima Daiichi accident. Since design inception of the facilities, systems were put in place which cope with the objectives of the four DiD levels relevant to design, even though the design of CNA I precedes those concepts.

The design improvements implemented after the Fukushima Daiichi accident have strengthen the safety of the units, having a main impact in level fourth and also level three of Defense in Depth. As examples it can be cited that level 1 of DiD has been reinforced by improvements of the availability of external power supply devises, etc., thus preventing initiating events; level 2 has been strengthen by the upgrading of safety related systems such as compressed air; level 3 is benefited by new or improved Emergency Power Supply systems, increased diesel generators autonomy, improved Emergency Water Supply, improved trip coverage, the addition of another Secured River Water Cooling pump, or improvements of the seismic capacity of the plants, etc.; and level 4 is further improved by the addition of several alternative water sources for both the reactor and the spent fuel pools, addition of Passive Auto-catalytic Recombiners, addition of Containment Filtered Venting systems, etc.

No. 14
Country: United Kingdom
CNS-REF.-ART.: General
PAGE OF REPORT: 8
CHAPTER OF NAT. Section 1.4.2.1.

SSR-2/1 Rev 1 (Paragraph 2.13) and the Fundamental Safety Principles in SF-1 (Paragraph 3.31) describes defence-in-depth and notes that the "independent effectiveness of the different levels of defence is a necessary element".
Can Argentina please provide more information on how it maintains the independence of the levels of defence-in-depth?
It is important to preserve the independence of the different levels of the Defence in Depth principle to assure the effectiveness of the measures within each level in the following terms:

- No measure can be avoided just by considering that another measure in a previous level was successful
- No measure can affect the application of another measure in a following level.
- As far as practicable, the systems intended for actuation at a certain level of DiD are not credited to play a role in another level. Whenever it is applicable, independence is reinforced by diversity and physical separation.
- It is recommended that different procedures are written for the operations in each level: normal operation, anticipated operational occurrences, accident condition, severe accident management and emergency plans.

Those criteria are applied at each safety review performed to the NPP.

No. 15
Country: Bulgaria
CNS-REF.-ART.: ARTICLE 6
PAGE OF REPORT: 31
CHAPTER OF NAT. Section 3.6.4.1.

The final report of PSA level 1 (high power) was submitted for approval to ARN in 2015 and it is being analyzed by ARN. The results of PSA L2 were also submitted for approval to ARN. Do ARN use the services of any technical support organizations for independent evaluation of licensees' safety assessments? According to the regulatory framework is there a time period for review and approval of documents submitted by the licensee?
ARN has a permanent technical staff for reviewing and evaluating of PSA levels 1, 2 and 3. The evaluation depth depends on the report, whether it is an updated or a new one.
In order to improve and complete some evaluations, ARN contracts TSOs (technical support organizations) for performing reviews. These TSOs have been GRS (Germany), INVAP (Argentina), independent experts in PSA, Sandia National Labs, among others.

The TSO reports are checked by the ARN technical staff, re-written if needed and addressed to the operator.

Regarding to the regulatory framework, ARN has a rule that state the schedule for submitting mandatory documentation to the regulator in the licensing process (AR 3.7.1). On the other side, the schedule links the regulatory review with a license or authorization milestone.

No. 16
Country: Canada
CNS-REF.-ART.: ARTICLE 6
PAGE OF REPORT: 42
CHAPTER OF NAT. Section 3.6.3.2.1.2.

It is not clear if the new emergency power supply system has already been installed? If yes, can you elaborate if the diesel generators are portable (on trailer) that can be transported in the event of a complete loss of onsite power – station blackout (SBO) or fixed on site?
The section referred in the question describes the design modification of the original Emergency Power Supply system of CNA I, replaced with a new, greatly improved system. The system is comprised of three stand-by diesel generators, which are not portable but fixed.

This new Emergency Power Supply system is already installed.

On a side note, in case of a complete loss of on-site power, there is a Mobile Diesel Generator as additional power source.

No. 17
Country: Canada
CNS-REF.-ART.: ARTICLE 6
PAGE OF REPORT: 34
CHAPTER OF NAT. Section 3.6.4.1.

Under the heading “SGs replacement”, the final paragraph states: “The design of the new SGs includes the following aspects: the re-powering of the plant…”. Please clarify what is meant by “re-powering of the plant”.
The SGs (Steam Generators) originally installed at CNE were manufactured with tubes of shorter length than those of other CANDU plants, making the total transfer surface 2800 m^2 and the thermal power transferred to the Secondary Steam System was 2015 MWth. Thus, during CNE refurbishment, SGs will be replaced considering a new design with longer tubes to achieve a larger transfer surface making the thermal power transferred to the Secondary Vapor System to reach 2064 MWth. With this and additional works in thermal cycle, turbine and alternator, CNE will increase output power from 648 MWe to 683 MWe.

No. 18
Country: Germany
CNS-REF.-ART.: ARTICLE 6
PAGE OF REPORT: 26
CHAPTER OF NAT. Section 3.6.3.2.1.2.

A new emergency power supply system was implemented at CNA I. It is stated, that the two redundant trains are diverse. Could Argentina explain in more detail how diversity was achieved to avoid a common cause / common mode failure of the emergency power supply system?
The referenced description mentions that the two EPS trains are diverse from other power sources, not diverse from each other. Nonetheless, actions have been taken to avoid common failures, as it is described in the referenced section of the report.

No. 19
Country: Germany
CNS-REF.-ART.: ARTICLE 6
PAGE OF REPORT: 28
CHAPTER OF NAT. Section 3.6.3.2.2.1.

It is described, that by a manual accident measure CNA II could be supplied via the 220 kV line from CNA I. Besides the benefits of such an accident measure, analyses of detrimental effects on safety are worthwhile to be performed. Can Argentina comment, if such analyses have been performed, before the decision was made to keep the connection between both Units?
As it was stated in the report, the interconnection referenced is neither automatic nor permanent, but has to be manually initiated by the operator in the event of an emergency power case with loos of diesel generators of one plant.

If such an event should occur, an assessment was performed in order to determine the load capacity that the interconnection could handle not jeopardizing the safety of the other unit. The interconnection would be seen from the CNA II electric systems as an external power source.

No. 20
Country: Germany
CNS-REF.-ART.: ARTICLE 6
PAGE OF REPORT: 28
CHAPTER OF NAT. Section 3.6.3.2.2.2.

Could Argentina discuss in more detail the measures foreseen in the new SAMGs SC 04-5 and SC 04-6 to increase the operating time of the diesel generators and the batteries, respectively?
As it was stated in the report, the interconnection referenced is neither automatic nor permanent, but has to be manually initiated by the operator in the event of an emergency power case with loos of diesel generators of one plant.
Due to the loss of external power supply (500 kV and 132 kV connections) and the impossibility of operating in island mode, the CNA II has 4 diesel generators of 6 MW each one (with 2oo4 redundancy), with an individual fuel consumption at 100% power of 1.47 m^3/h, the fuel availability

gives a range of 72 hours to the 4 generators. Taking out of service 2 of the 4 redundancies, the number of operating hours will be doubled.
The auxiliary boiler system has a tank of fuel with a volume of 200 m^3, adding 68 hours of autonomy for two of the generators. Fuel will be transferred from the auxiliary boiler system tank to the diesel tanks.
As a third measure, and if the interconnection between the auxiliary boiler system tank and diesel tanks is not available, fuel is supplied by a tanker.
These strategies will be used in case of any of the following plant conditions:

- Emergency diesel generators are the only source of electrical power.
- For external causes, the tanks of the diesel generators system can't be supplied with fuel.
- The external power supply to the site is expected to be interrupted for more than 72 hours.

SAMGs SC 04-6
The batteries are designed to maintain power supply to DC loads without interruption in the following cases:

- When the Auxiliary Power System fails and an additional single failure is assumed, for the power requirements supply of its associated train, and the power requirements of its loads feed through the decoupled diodes of its neighbor train, until the emergency diesel units take charge of the loads and the battery chargers are activated again (approximately 15 s) .
- When rectifiers are not available to supply the normal power of the 24 V and 220 V DC systems taking into account the power requirements of its associated train for 120 min.

The above design meets the minimum permissible load voltage, taking account of voltage drops (eg. cables, fuses, protectors and decoupling diodes).
The strategy to increase the maximum time of electrical supply of the batteries is the removal of consumption loads components are disconnected from different groups of the power supply system. These are detailed below:

1- Disconnection of consumers from 24/48 V power supply bars
The main components of the 24/48 V DC Emergency Power Supply System that are disconnected:

- Four 48/24 V DC buses, identified as [BVN/P/Q/R] Consumers disconnected.

2- Disconnection of rotary converters
The main components that are disconnected:

- Five rotary 220 V DC / 380 V AC converters,
- Four uninterruptible power supply 380/220 V AC

3- Disconnection of consumers in 220/380 V power supply bars
The 220 V DC Emergency Power main components that are disconnected:

- Four 220V DC bars, identified as [BVA/B/C/D] disconnected.

No. 21
Country: Germany
CNS-REF.-ART.: ARTICLE 6
PAGE OF REPORT: 31
CHAPTER OF NAT. Section 3.6.4.1.

For CNE a clause-by-clause comparison of the actual licensing and design basis with the most recent national and international safety standards was performed between 2013 and 2015. Can Argentina provide examples, which improvements have been identified by NA-SA and when the identified improvements will be implemented at CNE?
Many design changes were proposed by the designer taking into account the experience of other CANDU plants, and also by comparison with CNSC and CSA standards. As examples of the improvements we can mention:

Changes in EPS in order to support ECC.

Changes in EWS.

Changes in ECC design in order to accomplish single failure criterium.

Changes in SDS #1 (new shutdown parameters).

No. 22
Country: India
CNS-REF.-ART.: ARTICLE 6
PAGE OF REPORT: 32
CHAPTER OF NAT. Section 3.6.4.1.

It is stated "Addition of a line to add water in the calandria vault from outside of the reactor building. NA-SA submitted this design change to ARN for approval in 2014; it is still under ARN's analysis."
Does Argentina have plans for providing connection for addition of water in to calandria also from outside the reactor building to arrest the progression of the accident?
CNE is evaluating the details to implement a water line that allows adding water into the calandria, from a pump or from the network fire. This connection would be made in the pipeline of entry to the Moderator's Purification System.

Thus, during the CNE refurbishment, this design change will not be executed, but the connections necessary to be implemented after that will be provided.

No. 23
Country: Poland
CNS-REF.-ART.: ARTICLE 6
PAGE OF REPORT: 24
CHAPTER OF NAT. Section 3.6.3.1.1.

What is the definition of plant design modification? Is every change in any part of the plant regarded as modification?
Not every change done to the Nuclear Power Plant is regarded as a design modification, but only those that could alter the licensing basis. Nonetheless, not all design changes are treated equally from a regulatory stand point, but their treatment is graduated depending on the safety relevance of the modification.

The SAR describes the design features under which the facility is licensed, and hence any design change that alters such description, and consequently alters the licensing basis of the facility, ought to trigger the update of the SAR to reflect the design as built.

No. 24
Country: Poland
CNS-REF.-ART.: ARTICLE 6
PAGE OF REPORT: 31
CHAPTER OF NAT. Section 3.6.4.1.

Regarding Review and update of Hazard Assessments - is there a guide or other document explaining the classification of SSC into different change categories?
Yes, there is a document developed by the designer which defines the so called three categories: "like for like", "Minor Modifications" and "Major Modification or New Design". It also establishes the seismic hazard in terms of peak ground acceleration for each one of the categories.

No. 25
Country: Russian Federation
CNS-REF.-ART.: ARTICLE 6
PAGE OF REPORT: 29
CHAPTER OF NAT. Section 3.6.4.1.

From the report it is not clear the operating organization did not take part in the qualification tests of digital control computers?
Could you give additional information about operator and regulator participation in development of specifications (technical requirements) for Digital Control Computers; in design, engineering and factory testing of such computers.

It is not expected that qualification tests will be repeated at site. However NA-SA personnel witnessed these tests.
NA-SA participated as reviewer of the technical specifications and of the D&E phases and had active participation in the factory tests. These processes were also monitored by ARN.

No. 26
Country: Canada
CNS-REF.-ART.: ARTICLE 7
PAGE OF REPORT: 44
CHAPTER OF NAT. Section 3.7.3.3.

In the first paragraph, the report states “The Regulatory Body has performed assessments as well as… regulatory inspections and audits as frequently as considered necessary.” How is the scope of, and schedule for ARN inspections and assessments established? For example, is it defined on an annual basis?
The scope of the inspections and assessments is set by the Regulatory Body based on the following concepts.
It is considered that all objectives, requirements, conditions, provisions, pre-requisites, etc. that a nuclear operator must comply with to safely operate a facility, in the broadest sense of the concept, are adequately stated in the nuclear standards; in the Mandatory Documentation, such as Operating License, Handbook of Operating Policies and Principles, maintenance manuals, etc.; and in Regulatory Requirements issued by the Regulatory Body. Nuclear Standards might be reviewed over the course of years, in order to accommodate evolving world trends and new paradigms in the industry, safety developments, etc. Mandatory Documentation is often subject to review to reflect the evolving operating life and condition of facilities; and Regulatory Requirements are issued by the Regulatory Body every time that it is considered that action is required on the part of the Operator on a specific subject.
Based on the safety framework thusly described, the Regulatory Body develops a set of criteria, extracted from the instruments described above, for which fulfillment can effectively be verified by regulatory inspections and assessments.
On the base of those criteria, and in order to properly verify their fulfillment, the Regulatory Body develops a series of inspection tools, which are the activities effectively carried out by the regulatory inspectors and evaluators, and which yield a concrete result leading to the fulfillment, or lack of thereof, of the criteria. Such inspection tools are, for example, the review of the management of operative experience; oversight of the execution of periodic testing procedures; oversight of completion of maintenance programs; quality assurance audits; oversight of radiological protection programs results; oversight of ALARA program, etc.
Currently, the schedule of inspections is set taking into consideration the needed periodicity of each of those inspection tools in order to yield representative results of the areas assessed, and are scheduled on an annual inspection plan.
ARN is planning to elaborate a systematic inspection program (in terms of scope and frequency) for all NPP´s.

No. 27
Country: Austria
CNS-REF.-ART.: ARTICLE 7
PAGE OF REPORT: 40, 187
CHAPTER OF NAT. Section 3.7.2.2. and Section 4.6.

What is the envisaged timeline of the normative framework review that will incorporate the principles of the Vienna Declaration?
How does this activity and its timeline relate to Argentina’s intention to invite an IRRS mission and the timeline for that?
ARN has already started the review of their normative framework to incorporate the principles of the Vienna Declaration. The planned timeline to fulfill this objective is the end of 2019.
The visit of the IRRS mission to Argentina is planned to take place during the 4th Quarter of 2018. At that time, the review of the regulatory framework will still be in process and the progress achieved will be shared with the reviewers.

No. 28
Country: France
CNS-REF.-ART.: ARTICLE 7
PAGE OF REPORT: 44
CHAPTER OF NAT. Section 3.7.3.3.

Could Argentina describe with more details the daily tasks performed by resident inspectors at NPPs? What provisions are taken to ensure their independence from the operator (article 8.2 of the Convention)? For instance, do they work in a specific NPP for a limited period of time?
Question No. 26 refers to the scope and schedule of regulatory inspections in nuclear power plants. Please refer to the answer to that question for details on how the regulatory inspection plan is developed.
The daily tasks performed by the resident inspectors at NPPs consist on the execution of the inspection activities defined by the inspection plan, according to an annual schedule. This plan includes, among others, activities such as monitoring and control of plant parameters; regulatory oversight of maintenance programs, repetitive tests of safety systems, etc.; review of ALARA plan, doses; etc.
Other activities include the oversight of events, planned or unplanned power maneuvers, outages oversight, evaluation of emergency drills, etc.
In order to foster resident inspectors' independence from the operator, they receive a complete technical and material support from Regulatory Body headquarters. There are not regulations or limitations for the period of time to work in a specific NPP for inspectors, but at the present time, the average residence period in a specific NPP is 7 years.

No. 29
Country: Germany
CNS-REF.-ART.: ARTICLE 7
PAGE OF REPORT: 39
CHAPTER OF NAT. Section 3.7.2.1.

Does the new Regulatory Standard AR 10.10.1 define an exceedance frequency for hazards to be considered in the design basis? If yes, could Argentina share this value?
AR 10.10.1 does not define any exceedance frequency for external hazards to be considered in the design basis.

No. 30
Country: Germany
CNS-REF.-ART.: ARTICLE 7
PAGE OF REPORT: 43
CHAPTER OF NAT. Section 3.7.3.2.2.

In contrast to Atucha I and Embalse the licence for Atucha II is only valid for five instead of 10 years. Will ARN increase the validity period from five to ten years in the subsequent licences of Atucha II?
ARN will increase the validity period to ten years after successful completion of activities already requested to the utility and scheduled to be developed during the planned outages corresponding to years 2017 and 2018.

No. 31
Country: Mexico
CNS-REF.-ART.: ARTICLE 8
PAGE OF REPORT: 50
CHAPTER OF NAT. Section 3.8.3.

Currently, the ARN regulates three nuclear power plants in operation and the construction of a modular nuclear power plant among other safety-related activities. What provisions has the ARN (human and financial resources, and technical support organizations) to deal with the regulation of these nuclear power plants in the long term?
The regulatory system and practices are based on a graded approach following the safety classification of systems, structures and components. In performing the regulatory activities according with this approach, ARN has the practice to contract several TSOs (technical support organizations)

for some topical areas. For the core regulatory activities, ARN has hired professionals increasing the plant about 15% during the last year for dealing with NPPs. The plan is to hire professionals in order to preserve the knowledge of those that are going to retire during the following years.

No. 32
Country: United Kindgdom
CNS-REF.-ART.: ARTICLE 8
PAGE OF REPORT: 55-57
CHAPTER OF NAT. Section 3.8.4.

The regulatory body in Argentina, Autoridad Regulatoria Nuclear (ARN), has established, documented and implemented a Quality Management System (QMS) according to the requirements established in the 2015 version of the ISO 9001 Standard.

- **In 2015 the Board of Directors decided to continue the implementation, development and improvement of the QMS based on ISO 9001 Standard and its adaptation to the new version 2015 of that standard.**
- **The decision by the Board of Directors demonstrates continued commitment to an integrated management system that supports a strong safety culture within the regulatory body.**

Argentina appreciates the comment from United Kingdom.

No. 33
Country: United States of America
CNS-REF.-ART.: ARTICLE 8
PAGE OF REPORT: 59
CHAPTER OF NAT. Section 3.8.6.

ARN has continued to experience significant growth over the last three years in both its staffing and budget. Given that a significant portion of its budget comes from the National Treasury, what national motivation has driven this growth?

The national motivation was the National Nuclear Program that requires to increase necessary resources including ARN Budget to support fulfilment of regulatory activities, and enlarge ARN staff.

Accordingly, the main growth in budget from 2013 to 2015 is in the personnel item: from 115,781 to 170,811 (in thousands of Argentine pesos). In that period, ARN increased his staff from 396 to 449 people. Besides, inflation in the period must be considered.

No. 34
Country: Austria
CNS-REF.-ART.: ARTICLE 8
PAGE OF REPORT: 58
CHAPTER OF NAT. Section 3.8.4.5.

How many employees does the ARN Quality Management Unit have?

The National Report states on page 58 that 117 internal and external audits were carried out in the period 2013-2016 (August). That means more than one audit every second week. How does ARN Quality Management Unit cope with the (pre-, audit and) post-audit work: definition of action plans, monitoring actual implementation of recommendations, dissemination of the changes to the staff, monitoring of the effectiveness of these changes, and of the management system, etc.?

How does the rest of the ARN staff cope with the process, in terms of additional work load, adjustment to change and benefits from these improvements?

1) The ARN Quality management unit has 7 employees. All of them are Quality Internal Auditors and only 5 of them are quality leader auditor.

2) The Internal quality audits in the period 2013-2016 were 74 and the External Audits carried out by the IRAM (Argentine Institute for Standardization and Certification) were 43.
The quality management unit carries out an audit program that includes all the audits to be carried out during the year. Starting in 2016, this program is carried out every six months. This program is approved by the Boards of Directors and communicated to all ARN staff in their publication on the intranet.
The head of the quality management unit appoints a lead auditor and an audit team for each audit.

The lead auditor interacts with those responsible for the processes involved and sends the audit plan by mail. The audit team prepares the audit: reviewing applicable regulations, procedures, ISO 9001, etc.

Each Process is assigned a person responsible for implementing the quality management system. That person is a member of the quality management unit, who interacts with the process and carries out a control of the state of the implementations of the corrections and recommendations. Subsequently, the effectiveness of the implementation criteria is reviewed in the corresponding audit.

The processes that are certified with ISO 9001 carry indicators and quality objectives; they are reviewed and shown in the Boards of Directors; every suggestion of the Boards of Directors is made by mail to the corresponding Process.

3) The Board of Directors is informed of the audit findings states, proposals for future changes and the optimization of changes is analyzed to maximize benefits and reduce the increase of the work. ARN staff is notified by email and the person designated as implementer along with process owner decides the best way to implement the improvements.

No. 35
Country: China
CNS-REF.-ART.: ARTICLE 9
PAGE OF REPORT: 19
CHAPTER OF NAT. Section 2.13.

It is mentioned that "For the Second Hot Functional Plant Test (HFT II), the fuel assemblies were loaded in the Reactor, which is included in Phase B." However, Phase C was developed as a comprehensive unique Program for the whole Plant and is defined as Commissioning (meaning Nuclear Commissioning of the NPP).

Question: How to clearly divide the responsibilities of nuclear safety works between contractor and license holder in Phase B and Phase C? How to control the contractors effectively for license holder?

From the regulatory perspective, the responsibility is always under the organization that is the license holder. The activities performed in Phase B are covered by Construction License, instead of those under Phase C, for which Argentinean practice is to grant a Commissioning License. Regarding the so called “preliminary tests” performed under phase B and those corresponding to commissioning, AR 3.8.1 regulation establishes that the license holder is responsible for coordinating and controlling the development of all preliminary and commissioning tests, and they are allowed to delegate partial or totally the execution of them but retaining the whole responsibility.

No. 36
Country: United States of America
CNS-REF.-ART.: ARTICLE 9
PAGE OF REPORT: 65
CHAPTER OF NAT. Section 3.9.3.

In describing its fulfillment of Article 9, Responsibility of the Licensee Holder, ARN identifies that NA-SA carries out a Communications Annual Plan which has a stated goal “…to install a positive perception of the nuclear power generation.” This gives the impression that NA-SA’s efforts to promote nuclear power are a responsibility of the licensee imposed by ARN.

(1) Is this true?

(2) If yes, how does ARN maintain appropriate independence between promotion and regulation?

It is not true.

The comments regarding NA-SA´s activities in connection with public communication and stakeholders were indicated as a good practice in order to keep the public and governmental organizations informed about the facilities and activities. It illustrates a commitment regarding its obligations as an industry which offers a public service for the community in the provision of nuclear energy. However, this issue is not imposed by ARN. ARN is not in charge of the promotion of Nuclear Activities.

It is important to note here that the legislative framework from Argentina ensure real independence of the regulatory body from the industry. The National Law on Nuclear Activities Act No. 24,804 and its Decree No. 1,390/98 establish those provisions which are described in the following sections from the national report, such as;

1.2. NATIONAL POLICY IN THE NUCLEAR FIELD
3.7.1. NATIONAL LEGISLATIVE FRAMEWORK
3.7.3. LICENSING SYSTEM
3.8. ARTICLE 8: REGULATORY BODY
3.8.1. FUNCTIONS AND COMPETENCE OF THE REGULATORY BODY

No. 37
Country: Canada
CNS-REF.-ART.: ARTICLE 10
PAGE OF REPORT: 70
CHAPTER OF NAT. Section 3.10.2.2.

In the report, it is stated that Safety Culture is promoted based on diffusion, training, and re-training providing all personnel with the benefits of applying the safety culture principles to all activities carried out at NPPs.
How often and to what degree does the re-training take place for each employee?
The Safety Culture has a preponderant role in the training of all the staff.
All staff is annually trained. In 2014 the training was delivered to all the staff on Safety Culture features. In 2015, the results of a safety culture survey conducted at the plant were presented to all the staff.
In 2016, management expectations regarding Safety Culture were presented to all staff.
In 2017, it is planned to deliver a five-module training programme "Safety Start".

No. 38
Country: Canada
CNS-REF.-ART.: ARTICLE 10
PAGE OF REPORT: 71
CHAPTER OF NAT. Section 2.13.

It is stated in the report that "The programme defines management issues and implementation issues that require improvements. An eight step strategy for each topic is in place."
Can the Contracting Party elaborate more as to what the eight step strategy is?
As described in INPO 05-005, "Guidelines for Performance Improvement at Nuclear Power Station", 2006, the Eight Steps for Improving Nuclear Plant Performance are:
1. Establish Sense of Urgency
2. Align the Leadership Team
3. Develop or Revise Vision, Goals and Plans, Management Controls, and Performance Monitoring
4. Communicate the New Vision and Goals
5. Engage the Workforce for Broad Based Action
6. Create Short-Term Wins
7. Consolidate Gains and Produce More Change
8. Ingrain New Approaches in the Culture.

The framework for the steps and the sequence of implementation was derived from industry experience. The steps and the sequence have been reviewed in their final form by several of these executives and other leaders in the industry. The template general structure (process for change) is described by John P. Kotter of the Harvard Business School in his book Leading Change.

No. 39
Country: China
CNS-REF.-ART.: ARTICLE 11
PAGE OF REPORT: 16
CHAPTER OF NAT. Section 2.9.

CNA I and CNA II were initiated their commercial operation in 1974 and 2016. Their corresponding net electric powers are 335 MW and 693 MW. However, both plants use the same simulator for training, the differences between plant and full scope simulator of CNA II is probably to lead operators of CNA I to make wrong decisions.
Question: 1) How to evaluate the effectiveness of training on scope windows basis graphic simulator for CNA I? Is there any report about that? 2) Is the life extension for CNA I required by NA-SA and approved by ARN? If so, is the full scope simulator required?

1) The Interactive Graphical Simulator is used in the annual trainings for certain pre-established scenarios (it is a simulator of partial scope). The evaluation of the training is carried out by

observing the performance of the operators shift. There are no reports on the effectiveness of training.

From 1987 to 2015, CNA I Control Room staff was trained in the simulator of Angra II nuclear plant (PWR).

Atucha II Total Scope Simulator was installed on site in 2015.

Since 2015, CNA I Control Room staff has been trained in Atucha II (CNA II) simulator. Unit I and II designs are very similar so this has resulted in a great improvement in the simulator -plant similarity and in the applicability of emergency procedures.

2) The life extension of CNA I is under technical-economic evaluation by NASA authorities and a licensing framework agreement is under discussion with the Regulator. Up to now, the Full Scope Simulator is not required by the Regulator.

No. 40
Country: Canada
CNS-REF.-ART.: ARTICLE 12
PAGE OF REPORT: 78
CHAPTER OF NAT. Section 3.12.

The report describes “the measures taken by the Licensee regarding the contractors in order to ensure their adequate competence and safety culture.” Please describe the process used to vet contractor companies to ensure that their training and qualifications are adequate and contribute to the reduction of human error.

Before entering the plant, the contractor personnel receive training in human error prevention techniques. This initial training aims to present the human error prevention techniques, describes them, explains the importance of its use and gives examples comparable to the most common roles of contractors. In addition, those permanently hired participate in the "CAS" (Annual Safety Course), where topics such as human error prevention techniques and Safety Culture are presented.

The “Managers on the Field” program is based on task observation involving contractors and company staff. Managers observe the behaviour of contractors, as well as company supervisors and workers behaviours. Special focus is on supervisors setting expectations and correcting workers.

No. 41
Country: Canada
CNS-REF.-ART.: ARTICLE 12
PAGE OF REPORT: 79
CHAPTER OF NAT. Section 3.12.1.1.1.

Do you use observation and coaching techniques to reinforce desirable behaviors in the organization? If so, how would those techniques be incorporated into the other techniques that you use e.g., human error reduction tools?

There are two closely related programs focused on improving human performance and reinforcing desirable behaviours. These are: "Human Error Reduction Techniques" and "Managers on the Field". The “manager on the field” programme is based on task observation. Different activities are carried out to communicate and reinforce management expectations, such as:

- Annual re-training of task observers, to communicate expectations and show strengths and weaknesses of the processes.
- Weekly Reinforcement Messages. This is a weekly message focused on strengths and weaknesses of the organization on the application of Human Error Reduction Techniques
- Analysis of the Task Observation. Communications addressed to leaders taking into account actions to improve the task observation programme.
    - \- Periodic reports by each sector to identify adverse trends and taking action to reverse such trend.
    - -Compliance with the "Manager on the Field" and "Human Error Reduction Techniques” programs, are part of the General Behaviour Expectations of the Site Manager.

No. 42
Country: Romania
CNS-REF.-ART.: ARTICLE 12
PAGE OF REPORT: 78-84
CHAPTER OF NAT. Section 3.12.1./2./3./4.

What regulatory reviews and inspections are performed with a focus on human factors engineering and human performance? How does the ARN perform regulatory oversight of human factors?
Regulatory Oversight starts with Nuclear power plant personnel licensing (3.7.3.2.3.).
Besides, Operating Experience feedback follow up program is focused mainly on human factors engineering and human performance Operating Experience Feedback (1.4.2.2.3.).

No. 43
Country: Austria
CNS-REF.-ART.: ARTICLE 13
PAGE OF REPORT: 85-86
CHAPTER OF NAT. Section 3.13

According to section 3.8.4 the ARN is transitioning from ISO 9001:2008 to ISO 9001:2015, but section 3.13.2 states that the Quality Assurance programs of the organization units of the operating organization comply with ISO 9001:2000 and IAEA 50-C-Q (1996).
Is the ARN requiring the licensee to comply with more recent standards, especially with the view of refurbishment and life extension programmes?
If no, why not, and if yes, what is the timeline for the implementation of such requirement?
Currently NA-SA is assessing to adapt CNA U I-II and CNE Quality Assurance Manuals, according to ISO 9001:2015 and to IAEA GSR part 2 (2016).
ARN is preparing a new revision from AR 3.6.1 “Nuclear power plant quality system”, coherent with GSR part 2 and with other standards from the regulatory framework. The planned timeline to fulfill this objective is the end of 2019 (see question 27).

No. 44
Country: Bulgaria
CNS-REF.-ART.: ARTICLE 13
PAGE OF REPORT: 86
CHAPTER OF NAT. Section 3.13.2.1.

In para 3 is stated that: “These procedures foresee inspections at the supplier’s facility and review of their quality system before awarding the contract”. Are these procedures applicable to all suppliers in the country and abroad?
In order to qualify suppliers, a classification by safety classes is determined.
When the SSCs are Class I and the manufacturing is done in the country, inspections and audits are performed to the supplier.
When deemed appropriate, a permanent resident inspector remains at the factory.
When components are manufactured abroad, widely experienced suppliers are selected, recognized as suppliers of such components for the nuclear industry having international qualifications. In certain cases a third party can be delegated to audit or monitor on behalf of NA-SA during manufacture stage.

No. 45
Country: Canada
CNS-REF.-ART.: ARTICLE 13
PAGE OF REPORT: 85
CHAPTER OF NAT. Section 3.13.2

Can the Contracting Party explain if the quality program applies the graded approach or risk-based approach? Does the scope and frequency of an audit for a certain program use the graded approach in order to focus more on the higher risk areas?
Currently the quality assurance system is designed with the graded approach, but the organization is planning to address a risk based approach.
In the audit program, the graded approach is used and when the need for improvement is detected in an area, a surveillance program is implemented.

No. 46
Country: France
CNS-REF.-ART.: ARTICLE 13
PAGE OF REPORT: 87-88
CHAPTER OF NAT. Section 3.13.3.

Has ARN developed and implemented a construction inspection program to provide regulatory oversight of the construction of CAREM reactor? Does ARN carry out inspections at the construction site and/or conduct vendors inspections to ensure that products and services furnished to CAREM reactor meet established regulatory requirements for quality and other safety factors?
At present, ARN lead an inspection program at the site of the reactor CAREM 25 related with construction of civil structures (currently underway). This program is developed by ARN inspectors supported by experts belonging to some of the TSOs mentioned in National Report.
ARN reviews the mandatory documentation presented, in order to define and plan the inspection tasks (as were mentioned in National Report Section 3.7.3.3).
In order to meet established regulatory requirements, ARN only conduct inspections and audits to Responsible Entity, not to vendors. The Responsible Entity is in charge of assuring that the supplier quality program is implemented and the defined design criteria are accomplished.

No. 47
Country: Germany
CNS-REF.-ART.: ARTICLE 13
PAGE OF REPORT: 88
CHAPTER OF NAT. Section 3.13.3.1.

Section 3.13.3.1 deals with the quality assurance of subcontractors of the CAREM project. It is stated, that not only documents are verified, but also where responsible staff from the CAREM project will witness the tests. Could ARN share the involvement of the regulator in inspections on the vendors / subcontractors site in case of fabrication of items important to safety to ensure, that only qualified equipment will be installed at the site?
In order to meet established regulatory requirements, ARN only conduct inspections and audits to Responsible Entity, not to vendors. The Responsible Entity is in charge of assuring that the supplier quality program is implemented and the defined design criteria are accomplished.

No. 48
Country: Russian Federation
CNS-REF.-ART.: ARTICLE 13
PAGE OF REPORT: 85
CHAPTER OF NAT. Section 3.13.2.

According to the Report, hundreds of quality assurance procedures have been developed for each Argentina NPP in the framework of quality assurance programmes for these plants. Could you please give additional information about the subjects of these procedures and types of activities they address.
Do NPP quality assurance programmes include procedures for design / engineering companies, as well as companies testing systems and components supplied to nuclear power plants, in particular, in the framework of life extension projects (replacement /refurbishment of equipment)?
The quality assurance system is made of quality assurance programs.
There is one for each nuclear power plant, in which the works are described in documents that are categorized in:
Management: such as organization, policies, objectives, training of human resources, control of nonconformities, corrective and preventive actions, control of documents and records.
Operational: operation, maintenance, safety, engineering, production, human performance, training and development of personnel, technical assistance, site management, design, purchasing, inspection and acceptance tests.
Evaluation: self-evaluation and independent evaluation.
Quality programs audits are performed to selected suppliers in order to confirm that the supplier quality program is implemented and meets the established requirements. The main processes during manufacturing are also audited. When components are manufactured abroad, widely experienced suppliers are selected, recognized as suppliers of such components for the nuclear industry having international qualifications. In certain cases a third party can be delegated to audit or monitor on behalf of NA-SA during manufacture stage.

No. 49
Country: Canada
CNS-REF.-ART.: ARTICLE 14
PAGE OF REPORT: 94-100
CHAPTER OF NAT. Section 3.14.2.2.

Has the management of Central Nuclear Embalse pressure tubes and fuel channel components considered Operating Experience from other CANDU installations?

Within the framework of the CNE's Life Extension Project, fuel channels will be replaced as part of the task called "Retubing". Based on the experience of other CANDU-type reactors, the new components have design changes to improve aspects related to: plant availability, response to accidents, reduction of stresses in components, solution of operational experience events.

In the case of Pressure Tubes, changes in the chemical composition of the tube were implemented with the aim of decreasing its probability of failure. These changes consist of decreasing the initial hydrogen content from 20 ppm to only 5 ppm and limiting the chlorine content to 0.5 ppm as a residue of the process used to refine Zr in order to Improve material strength, performance against corrosion, reduce the amount of initial hydrogen which influences the phenomenon of DHC and blistering and Improve fracture toughness.

For Feeder tubes, changes in the chemical composition were implemented as well, in order to reduce the rate of thinning of material.

In the case of garter spring, it was implemented a change to a "tight fit" design, with greater capacity of adherence to the pressure tube and to use a material with greater elastic property and resistant to high temperatures.

No. 50
Country: Canada
CNS-REF.-ART.: ARTICLE 14
PAGE OF REPORT: 95
CHAPTER OF NAT. Section 3.14.2.2.1.

In the second paragraph (page 94), the report stresses the importance of repositioning garter springs “To verify… that the PT does not reach the content of equivalent hydrogen for the blister formation threshold (BFT)…”. BFT is the threshold for a contact blister to begin growing.

However, on page 95, the report states: “The general approach used… was to determine the probability of blister cracking…”.

Please clarify the operational limit NA-SA intends to apply when assessing its new fuel channels for contact: the pressure tube achieving BFT, or cracking of a contact blister?

On page 95, section 3.14.2.2.1 the report states: The general approach used in this evaluation was to determine the probability of blister cracking resulting from the failure process of the PT / CT contact causing blisters. Due to the fact that the blister cracking probability is depending of the time, it was considered 232,000 equivalent full power hours (EFPH) as the evaluation period, which exceeds the EOL and provides a margin for accounting the uncertainties in the repositioning activities.

The Annulus Spacer Position Study shows that no pressure tube to calandria tube contact is expected within the design lifetime (210,000 EFPH), with 87% margin. As such, from a design perspective, there is no operational limit to apply when assessing the new fuel channels, before the design life of 210,000 EFPH is reached.

Therefore, the operational limit that NASA applies is “the pressure tube achieving BFT”.

No. 51
Country: China
CNS-REF.-ART.: ARTICLE 14.
PAGE OF REPORT: 24
CHAPTER OF NAT. Section 3.6.3.1.1.

Description in section 3.6.3.1.1: "the SAR of nuclear installations must be updated each time that a plant design modification is performed".

Question: What are the requirements and principles on the update of the SAR for ARN?

The regulatory goal is to maintain the licensing bases documentation according to the actual and current plant configuration. This is required by AR 3.9.1 and the corresponding license.

As it was mentioned in the National Report, the mandatory documentation updating is carried out in NPPs based on the abnormal event evaluation, operating experience feedback, plant modelling with probabilistic techniques, identification of abnormal situations not specifically considered in the operation procedures, plant design modifications already implemented, etc. From this arises the need for the implementation of modifications or improvements.

No. 52
Country: China
CNS-REF.-ART.: ARTICLE 14.
PAGE OF REPORT: 30
CHAPTER OF NAT. Section 3.6.4.1.

Regarding CNE life extension activities, new systems have been added to provide redundancy during maintenance or upgrading of safety related systems.
Question: Could you please explain what kind of systems have been added, and what kind of features do those newly added systems have?
The systems that have been added and the kind of features that those newly added systems have, are described in pages 31, 32 and 33 of the section 3.6.4.1. Please, refer to them.

No. 53
Country: China
CNS-REF.-ART.: ARTICLE 14.
PAGE OF REPORT: 114
CHAPTER OF NAT. Section 3.14.3.3.

According to the IAEA Safety Standard Series No. SSG-25 "Periodic Safety Review of Nuclear Power Plants" document, 14 safety factors need to be reviewed in safety review activities. However the safety factor "Hazard analysis" wasn't included in the list of safety factors for CNA I PSR.
Question: Could you please explain the consideration for lack of the safety factor "Hazard analysis"?
IAEA Safety Standard Series SSG-25 “Periodic Safety Review of Nuclear Power Plants” in its item 2.12 states that: A PSR should provide a comprehensive assessment of the safety of the nuclear power plant. Since the complex process of conducting a PSR can be aided by appropriate subdivision of tasks, this Safety Guide sets out these tasks in accordance with 14 safety factors. These safety factors have been selected on the basis of international experience and are intended to cover all aspects important to the safety of an operating nuclear power plant. This subdivision is, however, not unique. In cases where the number of safety factors used and/or their grouping is different (for example, to meet the specific needs of the operating organization or regulatory body or owing to particular aspects of the nuclear power plant under review), the comprehensiveness of the PSR should be ensured by other means.
In the case of CNA I it was decided not to develop a Safety Factor “Hazard Analysis”; instead, the purposes of it were split and included in another safety factors as for example “Probabilistic Safety Analysis”. In this, all internal and external hazards are considered as contributions to Level 1 PSA.
Besides, regarding protection against external events like earthquakes, a seismic margin assessment was developed dealing with earthquakes of 10^{-4} frequency of exceedance per year.
In summary, a clause by clause comparison was performed covering the issues related with “Hazard Analysis”.

No. 54
Country: China
CNS-REF.-ART.: ARTICLE 14.
PAGE OF REPORT: 138
CHAPTER OF NAT. Section 3.17.2.3.1.

Question: CNA I and CNA II are comparatively old PHWR type. As to these two units, have them been implemented any significant design modifications according to the results of periodic safety review? What's the principle for determining these design modifications?
PSR was developed only for Unit 1. As a result of comparison with national and international standards, several improvements were decided. Some of them, as improvements in fire protective features, are being implemented. Also other design changes derived of Severe Accident Management Program, as PARs, were implemented in 2014.
Other major design changes, whose implementation demand a long outage of the plant, will be implemented as part of the plant life extension project.

No. 55
Country: Germany
CNS-REF.-ART.: ARTICLE 14.
PAGE OF REPORT: 103
CHAPTER OF NAT. Section 3.14.3.1.3.1.

Could Argentina inform about the scheduled date for the installation of the filtered containment venting system at Atucha I? Are there similar plans for Atucha II?
Starting from the simulation of the Plant behaviour in case of Design Extended Condition scenarios without countermeasures (base cases) using the programs: MELCOR (Atucha II) and RELAP5/SCDAP (Atucha I and Atucha II), insights have been gained into the PHWR severe accident issue.
Diverse Severe Accident Management countermeasures are presently being assessed for Atucha I and Atucha II.
They shall take into account the design particularities of those Plants, both inside and outside of the reactor pressure vessel.
Some features, like the moderator tank and the lower plenum filling bodies, have a significant role in the in-vessel accident progression.
On the other hand, the reactor pressure vessel cavity has also some unique characteristics that require a specific approach to evaluate countermeasures like external reactor vessel cooling (ERVC).
The ERVC efficacy to stabilize the corium inside the vessel is currently being analysed.
Preliminary analysis considering a Filtered Containment Venting System (FCVS) for Atucha I and Atucha II were performed last year (for instance simulations with the GOTHIC code).
It was concluded that more knowledge about the Plant behaviour is needed.
Depending on the ERVC evaluation results and other considerations, the analysis of FCVS for both Plants is planned to carry on.

No. 56
Country: Germany
CNS-REF.-ART.: Article 14.
PAGE OF REPORT: 110-114
CHAPTER OF NAT. Section 3.14.3.2.

External events could also contribute to the core damage frequency. Does Argentina have plans to expand the scope of probabilistic safety analyses in such a way, that also external events will be taken into account?
As were mentioned in the National Report, the most significant external hazards affecting the Argentine NPPs are earthquakes; flooding and low level water.
“For Argentinian NNPs, a PSA based Seismic Margin assessment (SMA) was performed to determine potential vulnerabilities of the current design to face seismic event that could jeopardize the fundamental safety functions. To perform the seismic safety assessment (SSA), the Licensee of CNA I; CNA II and CNE updated the seismic hazard of each site using methodologies and databases according to the state of the art.”
In the case of Flooding and low level water for Atucha site:
“The maximum and minimum water levels in Atucha I-II site are being reassessed. The following scenarios are being considered:

1) For maximum level rise: a chain breaks of Itaipú and Yacyretá, in simultaneous with a maximum Paraguay river flow, maximum rainfall on Paraná river basin and extreme rise of Río de La Plata river. In all cases historical records were taken as reference.
2) For minimum water level: a minimum Paraná flow river with a 100 years recurrence in simultaneous with the minimum historical level in Paraná de Las Palmas discharge point.

The following studies have been completed so far:

- General description of the area of study.
- Weather and meteorological events in the area where Atucha site is located.
- Hydrodynamic Model".

Besides, "there was a reassessment of the risk of tornadoes on Atucha site, which also evaluated the impact of missiles in CNA I buildings.

The report elaborated gathers and evaluates national and international standards applicable to missile impact generated by tornadoes. A study was conducted of the physical characteristics of tornadoes, storms and missiles that could impact on the facilities; a probabilistic model on tornado risk was developed and applied in order to determine exceedance curves and the return period of wind speeds caused by tornadoes and other severe storms in the site area. For this reassessment, a record of tornadoes spanning until 2013 was used. Finally, the general condition of each building of CNA I regarding safety against missile impact was detailed.

Regarding intense rains, lightning and tornadoes, the conclusion of the assessments was that a suitable margin exists and some conceivable weaknesses were identified as well as some improvements and modifications were proposed. The licensee decided to implement additional studies to confirm them".

Flooding / low water level for CNE:

"The consideration of flooding/low-water-level for Argentine NPPs is consistent with both domestic regulations and international criteria established at the design time. However, further studies were considered necessary to be required for the Embalse site. For CNE, it is foreseen to carry out a re-evaluation of the consequences of the occurrence of earthquakes on the existing dam located downstream from the plant.

- Internal and external flooding situations have been analyzed. Regarding this issue, it is considered that the Licensee is carrying out appropriate actions to successfully cope with these scenarios".

"Tornadoes, wind loads, lightning and intense rains have been analyzed and it is considered that the licensee is carrying out the appropriate actions to successfully cope with these scenarios".

"A re-evaluation of the risk of tornadoes for the Embalse site is foreseen to be completed by the end of 2017. This re-evaluation will include the response analysis and the existing margin for the safety related buildings and SSCs facing Beyond Design Basis (BDB) tornadoes. The effect of missiles caused by tornadoes will also be included".

In the future, ARN will promote the realization of PSA including external events.

No. 57
Country: Romania
CNS-REF.-ART.: ARTICLE 14.
PAGE OF REPORT: 4
CHAPTER OF NAT. Annex II

The use of PSA in the CNE´s Life Extension (PLEX) program is mentioned as a good practice. Please provide information on how was the PSA used in the PLEX program.

Starting from the simulation of the Plant behaviour in case of Design Extended Condition scenarios without countermeasures (base cases) using the programs: MELCOR (Atucha II) and RELAP5/SCDAP (Atucha I and Atucha II), insights have been gained into the PHWR severe accident issue.

The PSA was a tool to verify adequacy and appropriateness of the modifications. The two main issues which relate PLEX program to PSA are:

1) The use of PSA results as a tool to contribute in the verification of the design changes to be carried out during the Refurbishment Project.
2) Updating of the PSA model, including all the design changes carried out during the Refurbishment Project.

Regarding 1), the main Minimal Cut Sets contributing to the Core Damage Frequency were evaluated. Some design changes developed during the Refurbishment Project were taken from this analysis in order to improve systems capabilities or redundancies and eliminating human actions through the automation of actions, etc. Some examples of them are the following:

- Generation of a LOCA signal on sustained PHTS header low pressure, in order to automate the injection of the ECCS for Very Small LOCA scenarios. This injection was manual before the Refurbishment activities.
- Automatic stop of the four PHTS main pumps on low PHTS header pressure (LOCA Scenario) to avoid PHTS additional damages, not depending on operator actions.
- For the same reason that the aforementioned, automatic stop of the four PHTS main pumps on high pump bearing temperature (loss of pumps cooling because of a loss of service water).
- Automatic alignment and actuation of the Low Pressure ECCS stage in order to avoid human actions.
- Addition of redundancy for the ECCS valves that isolate dousing tank at the end of the ECCS Medium Pressure actuation in order to improve the reliability of the safety function “avoiding air intake to PHTS” to assure the circulation.
- Addition of redundancy for EWS main valves 3461-PV7 and PV41 to diminish the probability of failure to open for the injection from dousing or from the lake.

Regarding 2), regardless of the design changes that are more relevant from PSA point of view, other design changes are being carried out. All design changes affecting to the PSA model were included in such model in order to keep PSA results updated.

No. 58
Country: Russian federation
CNS-REF.-ART.: ARTICLE 14
PAGE OF REPORT: 187
CHAPTER OF NAT. Section 4.4.

According to the Report, the Licensee has developed a Programme of Consolidation of Safety Culture, the performance of which is measured through indicators.
Could you please give examples of such indicators, and tell whether you use quantitative characteristics to evaluate safety culture?
Establishing a quantitative measure using indicators on Safety Culture is not straightforward to do. However, it is very useful and necessary to have indicators that show some trend in issues concerning to Safety Culture. For this reason, some indicators have been defined in relation with some attributes of the traits of a Healthy Nuclear Safety Culture. For example:

- PRACS-002 Indicator: Questioning attitude. Number of Level 4 events reported in a quarter vs. the total number of events reported in the period.
- PRACS-004 Indicator: Leadership Accountability. Total number of tasks observations performed by quarter and by Deputy Manager, Department Head and Division Head vs. the number of observations expected.
- PRACS-007 Indicator: Continuous Learning. Number of people performing Benchmarking, WANO Missions, Workshops, Internal Technical Support Missions (MISTI).

No. 59
Country: Russian federation
CNS-REF.-ART.: ARTICLE 14
PAGE OF REPORT: 101
CHAPTER OF NAT. Section 3.14.3.1.3.

The Report mentions some technical tools that have been introduced (or are planned to be introduced) at Argentinian plants to manage beyond-design-basis accidents.
Could you please tell whether these tools (in particular, hydrogen recombiners) will undergo periodic checks, including testing, throughout the plant lifetime?
Autocatalytic hydrogen recombiners are installed on site and are periodically tested during scheduled outages.
According to AREVA recommendations, at least 25% of recombiners must be tested at each scheduled outage.

No. 60
Country: Canada
CNS-REF.-ART.: ARTICLE 15
PAGE OF REPORT: 119
CHAPTER OF NAT. Section 3.15.4.1.

In Table 3.15.7, it seems that the I-131 release for 2015 is many orders of magnitude better than 2013/2014 and significantly better than the other nuclear stations. This seems to be not proportional with the trend / magnitude of the other emissions. Can the Contracting Party comment on this value and confirm that it is correct?
Embalse NPP has operated during 2015 only in July and some days in December. The releases of gaseous I-131, noble gases and aerosols were not detectable or barely over the detection limit during those periods. The values for aerosols and Xe-133 in the Table 3.15.7 are due to the emissions registered in July but the I-131 was hardly detected.

No. 61
Country: Canada
CNS-REF.-ART.: ARTICLE 15
PAGE OF REPORT: 120
CHAPTER OF NAT. Section 3.15.4.2.

This section documents a loss of heavy water from the steam generator of Central Nuclear Embalse. Can the Contracting Party explain how the event happened, and discuss any corrective actions put in place to prevent reoccurrence?
The contribution described in section 3.15.4.2 from tritium release from Embalse NPP was not produced by the occurrence of an event, but from a small leak from a tube of one of the steam generators, whether from surface pitting or from the welded connection to the tube plate. Embalse NPP has in place a procedure for detection and repair of leaks from steam generators tubes, but the successful application of the procedure requires the leak rate to reach a certain minimum value in order to provide relevant possibilities of detecting the offending tube. In the referenced case, the leak rate remained minutely low during an extended period of time, which is contrary to what common operative experience shows, thus preventing the possibility of successful discovery of the leaking point. During plant outages, inspections were carried out over the steam generator, and tubes were sealed on the smallest of indications, but the leak remained.
Nonetheless, it is to be noted that the dose to the representative person, even in this situation, remained orders of magnitude below regulatory limits.
Corrective actions put in place to prevent reoccurrence consist of the replacement of all four steam generators of Embalse NPP.

No. 62
Country: Canada
CNS-REF.-ART.: ARTICLE 15
PAGE OF REPORT: 122
CHAPTER OF NAT. Section 3.15.6.

The report details some interesting dose savings achieved for the Reactor Inspection Program tasks. The notable improvements in collective dose per related tasks should be shared through operational experience processes to ensure others can learn from these successes.
Does Argentina have mechanisms in place and regulatory obligations to communicate lessons learned? Is Argentina an active member of the NEA's ISOE program?
Convention report is a mechanism to communicate lessons learned (Annex V).
Periodically, incidents significant to safety are reported by both CNA (CNA **I-II**) and CNE to the WANO. In addition, CNE usually participates in CANDU Owners Group (COG) Weekly Screening Meeting teleconferences, where it provides to COG a periodic report of its significant events.
Currently Argentina does not belong to OECD / NEA.

No. 63
Country: China
CNS-REF.-ART.: ARTICLE 15
PAGE OF REPORT: 125
CHAPTER OF NAT. Section 3.15.6.3.

One application of ALARA practices presents the Digital RWP implementation.
Question: What's the difference between former RWP and present digital RWP, and how the new method benefits ALARA practice of onsite radiological work?
The difference between the current RWP (Radiological Working Permission) and former one is based on a digital system developed to easily manage all permits (previously were all written in paper). The benefit is not only for all CNE personnel as they are now able to track the RWP status, but for ALARA officers also to have more flexibility and facility in providing the service.

No. 64
Country: Canada
CNS-REF.-ART.: ARTICLE 16
PAGE OF REPORT: 131
CHAPTER OF NAT. Section 3.16.6.

It is stated that the "municipal plans are tested at least once per year, involving response organizations and the public." How do you get the public involved in the exercises? Are there a minimum number of public participants for each exercise?
Public located in the surroundings of NPPs is encouraged to participate voluntarily in exercises by training in a realistic nuclear emergency situation, where several protective actions are to be implemented promptly. Among response actions to be trained, evacuation (within an area of 3 km from the NPP), sheltering, iodine thyroid blocking and restrictions for accessing to the affected zone are included. A few weeks prior, public meetings and presentations on educational institutions are organized to provide information about the exercise, but also to retrieve feedback from the public.
During the exercise, the public is notified about a nuclear emergency situation through the Public Alert System, which consists of a set of sirens distributed across the settlements close to the NPP. As further indications are delivered through local FM stations, people are encouraged to stay tuned at every stage of the exercise. Also, a candy (emulating the stable iodine) is provided to each member of every home by Gendarmería Nacional Argentina.
At some point, local FM stations will indicate that it´s time to implement sheltering and to take the stable iodine and people are expected to do so. The experience showed that communication through local FM stations is an effective channel for giving indications to the public, and has also proven to be a useful way to obtain feedback as people are animated to leave messages and inquiries to be answered by a designated officer. Implementation of almost every protective action has been improved by taking into account people´s outlooks.

No. 65
Country: United States of America
CNS-REF.-ART.: ARTICLE 16
PAGE OF REPORT: 127
CHAPTER OF NAT. Section 3.16.2.

The National Report indicates that the regulatory body will take charge of emergency management in the event of a nuclear accident. In other countries, the regulatory body serves in an advisory role to the licensee and national and local governments to maintain its independence.

1) **Please elaborate on what lessons learned prompted the Argentinian regulatory body to assume this role.**
2) **What are the benefits and challenges with such an approach?**

The main strength of this approach is that the regulatory body as the party in charge of emergency management has the sufficient knowledge on nuclear safety, due to its regulatory functions, to implement effective and on-time protective actions, but also to modify these response actions, in case it was necessary. The regulatory body has also adequate monitoring capabilities and a great experience in radiological protection issues.

Particularly, the regulatory body in Argentina has also the responsibility to manage radiological emergencies. Therefore, the regulatory body should have enough resources and infrastructure to face this duty.

One of the challenges of this scheme is that coordination with other response organizations may result complicated, but it was overcome over time.

Ultimately, it was settled that for our country the regulatory body being in charge of the emergency management has more benefits than challenges.

No. 66
Country: India
CNS-REF.-ART.: ARTICLE 16
PAGE OF REPORT: 135
CHAPTER OF NAT. Section 3.16.10.2.1.

It is stated "A new Internal Centre for Emergency Control was built. The Centre was designed and built taking into account the recommendations and suggestions provided by WANO in the 2012 Peer Review and in the Emergency Preparedness Technical Support Mission that took place in 2013, as well as the proposals that came from performed benchmarking."

What is the design basis of the Internal Emergency Control Centre located at site with regard to seismic, flood and radiation protection considerations and what are the main plant parameters monitoring and power supply provisions available in the Internal Emergency Control Centre?

The original design criteria of the “Internal Emergency Control Centre" was not based on seismic conditions or flooding because it is located in a "non-seismic" area and at a height of 23 m above the level of the river. Nevertheless, NA-SA is currently working on a new design based on these external events.

In the current building a ventilation system with external air filtration (HEPA absolute filter and activated carbon) is being implemented to reduce radioactive material from the outside.

The Centre has 3 redundant and diverse electrical feeds.

Regarding "main plant parameters monitoring", the building has optical fiber connector to access documentation servers and plant parameters of both NPPs. Also the building has a back-up digital data server. In this way, NA-SA has access to all the information concerning reactors.

The Internal Emergency Control Centre has redundant communications: land line phones, mobile phones and satellite phones and VHF radio.

In the Internal Emergency Control Centre, on-line information is received from the perimeter dose rate measurement system within a 10 km radius around CNA site. It has its own meteorological tower that provides information such as wind speed and direction, solar incidence, rain gauge, etc.

In case of emergency there is a loudspeaker system installed within a 10 km radius around CNA site to alert the population.

There are potable water and packaged food for the emergency organization and exclusive dosimeters to be distributed among the Emergency Response personnel (Police, Firefighters, Civil Defence, etc.).

No. 67
Country: United Kingdom
CNS-REF.-ART.: Article 16
PAGE OF REPORT: 130
CHAPTER OF NAT. Section3.16.5.

The report states that during an emergency, relevant automatic protective actions are taken within the Protection Action Zone (PAZ) and Urgent Protective Zone (UPZ) and that "Environmental monitoring starts once the release of radioactive material has finished." Given the new real-time information environmental monitoring network of fixed and mobile radiological and meteorological stations around each plant.
Please describe any plans the Nuclear Regulatory Authority (Autoridad Regulatoria Nuclear (ARN)) has to integrate this and other field monitoring data into decision making processes for off-site protective actions prior to termination of the release.
To this day, there is no integration of monitoring data obtained from fixed stations or mobile instruments with the decision making process for off-site protective actions prior to termination of release. Urgent protective actions within the PAZ are implemented automatically once the emergency has been declared.
For the moment, information from fixed and mobile monitoring is part of the data taken into account to implement and to adjust protective actions once the release has finished, for example, adjusting the extent of the PAZ and as data entries for atmospheric dispersion models.

No. 68
Country: Canada
CNS-REF.-ART.: ARTICLE 17
PAGE OF REPORT: 141 and 144
CHAPTER OF NAT. Section 3.17.2.3.2.1.

Intake channel blockage has been studied as a result of flooding (page 141). However, it is not clear if these studies have addressed the extreme scenario of the rupture of the upstream Yacyretá Dam and consequence of extremely large amounts of debris, including whole trees, on the potential blockage of the intake channel.
For the case of Atucha site (CNA I and CNA II), the cooling water is extracted from the Paraná River through the water intake. Water is filtered through thick, thin grids and rotating filters.
The rupture of the upstream Yacyretá Dam has been considered and the specific information is included In the Seventh National Report, item 3.17.2.3.3.1.2 (Flooding/low water level). In that item it is mentioned that "the estimated maximum water height that would be reached on the Atucha site after the rupture of the Yacyretá dam located 1,200 kilometers upstream is similar to the one calculated for the simultaneous confluence of the two main river tributaries in case of a maximum precipitation (intense rains). For the dam rupture above mentioned, it was estimated that the probable maximum high water level (PMH) for the Atucha site is 8.45 meters…"
Besides, it is explained that "the CNA I and CNA II main buildings were built on a 23 meters height plateau. Therefore, high water levels are not expected to affect these buildings since those are all placed at a level that provides a substantial margin from the PMH.
However, in the CNA I case, the water intake for the normal river water cooling system pumps and the secured river water cooling system pumps are located at a level of 6 meters and may be vulnerable to river level rise for it is less than the PMH of 8.45 meters. Besides, the minimum water level height for which the plant can operate is –1 meter given by the pumps water intake level.
To overcome these CNA I extreme high and low water levels design weaknesses, a fourth pump of the secured river water cooling system (UK) was installed in the CNA II pumps house. This pump keeps running even with a river level rise of 8.45 meters or low river levels of -2.00 meters. This allows this pump operation even in case the water level exceeds the CNA I pump house height or the water level is below the pump intake level".
On the other side, "the Second Heat Sink (SHS) system belonging to CNA I NPP is capable of removing heat in situations where the pump house is unavailable.
Also the availability of the structures, systems and components (SSCs) that must be functional in case of external flooding scenarios has been studied. The SSCs that are necessary to stop, bring and maintain the reactor into a safe shutdown condition after a loss of the river water assured cooling system (UK), were identified from the plant's PSA as well as their availability was verified through the plant walk-downs".

As well, in the same item it is mentioned that “CNA I and CNA II have three specific flood and low water level management instructions. In case of an extreme scenario like the one described before, the NPP will be kept in a safe shutdown state”.

No. 69
Country: Canada
CNS-REF.-ART.: ARTICLE 17
PAGE OF REPORT: 137
CHAPTER OF NAT. Section 3.17.2.

The discussion of site re-evaluation post-Fukushima is comprehensive and indicates that a thorough review was performed.
Has the capability of the facilities to manage radioactive liquid waste (i.e., contaminated cooling water) following a core damage event been assessed and/or evaluated?
The radioactive liquid waste generated as a result of a severe accident shall be kept inside the containment. This is possible due to the big containment volume that is similar to PWR. The indicative chart of the different contributions is attached.

Acronyms:
RB: Reactor Building
HTS: Heat Transport System
HP ECC: High Pressure Emergency Core Cooling
MP ECC: Medium Pressure Emergency Core Cooling

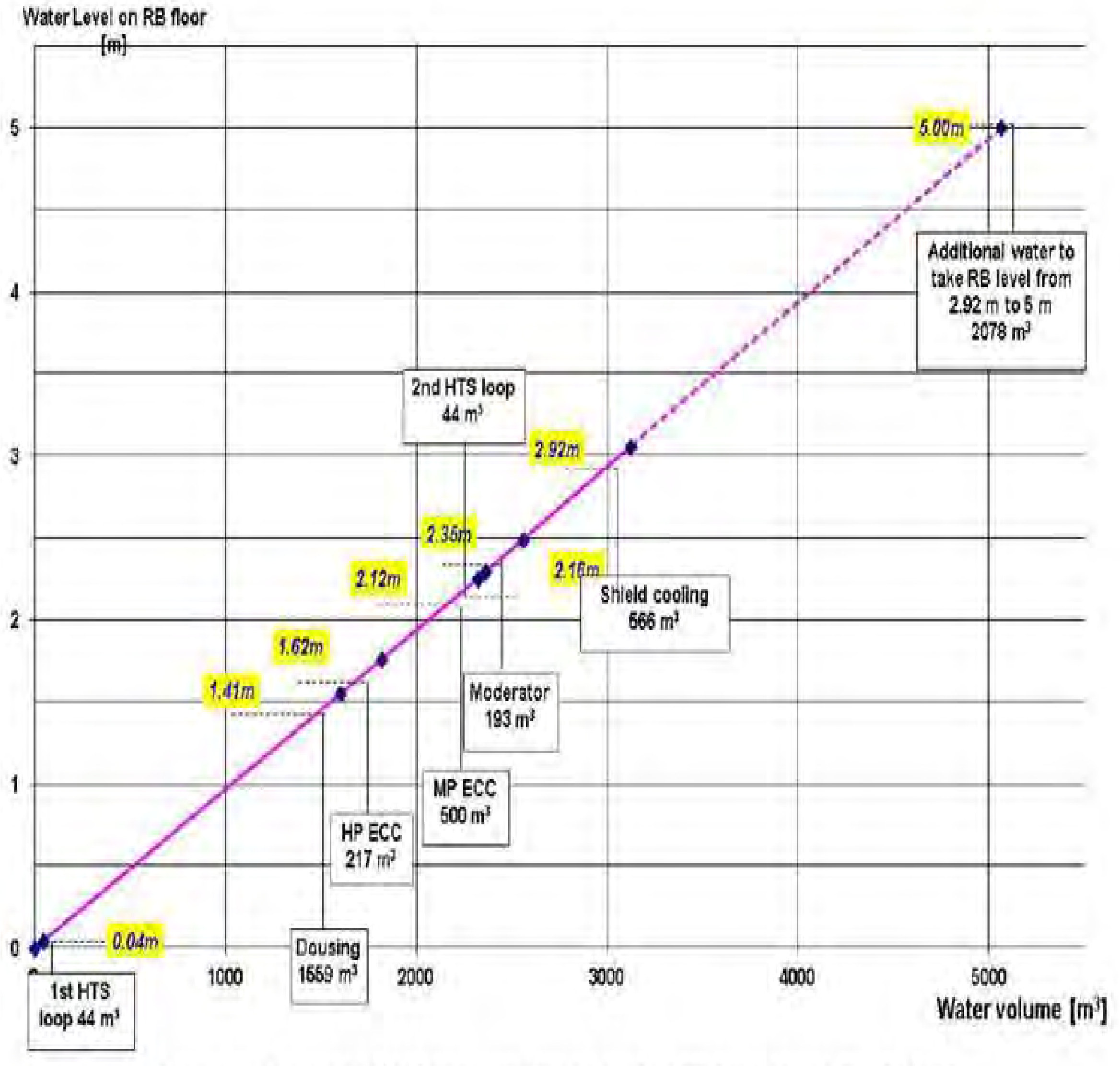


Figure 1 – Embalse Containment Water Level vs Water Volume [Up to 5 m]

No. 70
Country: Canada
CNS-REF.-ART.: ARTICLE 17
PAGE OF REPORT: 156
CHAPTER OF NAT. Section 3.17.2.3.2.

The site re-evaluation summary highlights ends with the following statement: "as a result of the stress tests performed by the Licensee and the corresponding assessment carried out by the Regulatory Body, it was issued a regulatory requirement asking additional assessments or the implementation of improvements and modifications referred to seismicity, flooding and other extreme external hazards, many of which have already been implemented".
It is commendable for Argentina to take the necessary steps to implement these improvements.
What is the timeframe for licensees to complete these implementations?
In sections 3.14.3.1.2. Post Fukushima Analysis and 3.18.3.2.1/2, 3.18.3.3.1 Design improvements implementation, the improvements and modifications referred to extreme external hazards among other plants changes are stated. Several changes have been already implemented; however, there are few improvements still under analysis.
In order to define a time frame it is necessary to finalize those analyses and verify the feasibility of such remaining improvements.

No. 71
Country: Germany
CNS-REF.-ART.: ARTICLE 17
PAGE OF REPORT: 138
CHAPTER OF NAT. Section3.17.2.3.

In the report it is said, that the seismic risk at the Atucha site is very low. And the NPP shall withstand horizontal loads typically for regions with low seismicity. Which PGA value is used for the design basis earthquake for Atucha I and Atucha II? Does Argentina apply the minimum PGA value of 0.1g as recommended by the IAEA Safety Standard No. NS-G-1.6?
As stated in section 3.17.2.3.2.1.1. "The CNA I NPP was not originally designed or qualified considering severe earthquakes. However, due to the conservative design applied as well as the SSCs robustness, it was considered that there is an inherent capability to withstand earthquakes of a certain level which will be determined by means of a SMA to assess the SSCs' status in relation to their ability to perform its safety function after a specific earthquake occurrence. The CNA II original design criteria were based on a 0.05 g PGA DBE (Design Basis Earthquake). Additionally, design principles and construction measures for low seismicity regions were applied".
A hazard analysis for Atucha site was performed, demonstrating that a 0.1g PGA value must be considered as design basis earthquake for new NPPs and as review level earthquake for existing NPPs. This is in line with IAEA Safety Standard No. NS-G-1.6.

No. 72
Country: United Kingdom
CNS-REF.-ART.: ARTICLE 17
PAGE OF REPORT: 137
CHAPTER OF NAT. Section 3.17.2.1.

The report states that the information provided by siting studies enable the identification of actions required to protect the public from accident situations, which can then be taken into account within the corresponding Emergency Plans. An increase in population around nuclear sites has the potential to affect the safety of both individuals (by potentially decreasing the protection that is provided by the provisions within the off-site plan) and society (by increasing the collective effective dose that the public would receive in the event of an off-site release of radioactive material).
Please clarify what regulatory or other administrative controls are in place to limit the population distribution or growth around a site that may be considered as suitable for the location of a nuclear power plant?
The regulatory body or the licensee does not have authority to control or to limit the population distribution, the population growth or the long term land-use policies and plans in the regions surrounding a nuclear facility.
Nevertheless, AR 10.10.1 standard "Sitting assessment for Nuclear Power Plants" may be considered as a tool that influences indirectly in this subject. One of the requirements stands that "Population growth and distribution must be kept under surveillance […] in order to keep associated risks as low as reasonably achievable", but also that "Sitting assessment must consider expected variations on land usage in the surroundings of a NPP".
In the section related to the public and emergency preparedness and response, the standard states that "The geographic area surrounding a NPP shall be considered taking into account present and future population characteristics and distribution, including present and future land and water usage, but also any other characteristic that can be affected or can affect the consequences on the public or the environment of radioactive releases."

No. 73
Country: Canada
CNS-REF.-ART.: Article 18
PAGE OF REPORT: 157
CHAPTER OF NAT. Section 3.18.3.2.1.

In the "Review of Safety Issues", the positive void coefficient is mentioned as an issue for Central Nuclear Atucha II, but not for Central Nuclear Atucha I or for Central Nuclear Embalse.
Can the Contracting Party explain the rationale for this?
The "Review of safety issues" was included as an explanation of the additional evaluations of the SAR required by ARN for the re-licensing of CNA II after several years of delay, with the objective of reevaluating and reinforcing the design basis of the plant and safety systems. This imply that the positive void coefficient is not a safety issue for any CANDU reactor type nuclear power plant or for CNA I (both PHWR reactors).

No. 74
Country: China
CNS-REF.-ART.: Article 18
PAGE OF REPORT: 139
CHAPTER OF NAT. Section 3.17.2.3.1.

Description in section 3.17.2.3.1: "Man induced external events are also considered in the plant design basis. In this regard, an explosion pressure wave corresponding to deflagration of a gas cloud is defined, which is duly taken into account through appropriate layout and structural aspects. Also, measures to avoid the entering of explosive gases into buildings and structures are taken into account in the design of the plant".
Question: How to take into account the possibility and corresponding measures for hydrogen explosion/deflagration inside the reactor building?
As were mentioned in the previous report related to the potential hydrogen explosions, in CNA I and CNA II Passive Auto-catalytic Recombiners (PARs) have been installed and in the case of CNE, PARs installation required for the hydrogen management inside the containment is foreseen to be implemented during the life extension refurbishment.

No. 75
Country: Germany
CNS-REF.-ART.: ARTICLE 18
PAGE OF REPORT: 162
CHAPTER OF NAT. Section 3.18.3.4.5.

A high quality of the reactor pressure vessel is a mandatory requirement for nuclear safety. One issue is, that the reactor pressure vessel counts to the so called long lead items. Regulatory inspection of the fabrication may be challenging from a legal point of view. Could ARN describe in more details the planned inspections of the reactor pressure vessel during fabrication to ensure a high quality of this component?
The CAREM RPV is designed and constructed under ASME III stamp, in accordance with the relevant requirements of Section III of the ASME Boiler and Pressure Vessel Code, 2010 Edition with 2011 Addenda.
ARN reviews the mandatory documentation presented, in order to define and plan the inspection tasks (as were mentioned in National Report Section 3.7.3.3).
In order to meet established regulatory requirements, ARN only conduct inspections and audits to Responsible Entity, not to vendors. The Responsible Entity is in charge of assuring that the supplier quality program is implemented and the defined design criteria are accomplished.
Related to MIP (Manufacturing Inspection Program and Testing), the Responsible Entity establish, by the Design Specification of CAREM RPV, all the examinations, dimensional controls and tests that manufacturer shall be responsible of performing (under ASME NB-5000 and NB-6000). All the examinations shall meet the written procedures which have been previously accepted by the Responsible Entity. The examinations and tests shall be subject to Responsible Entity acceptance.

No. 76
Country: Canada
CNS-REF.-ART.: ARTICLE 19
PAGE OF REPORT: 101, 156 and 159
CHAPTER OF NAT. Sections 3.14.3.1.2., 3.18.3.2.1./2. and 3.18.3.3.1.

In response to the Fukushima accident a significant safety analysis appears to have been completed to determine the required mitigating actions, subdividing actions between short-, medium- and long-term plans.
However, much of the corrective actions are to be implemented as part of the long-term plan. Can the Contracting Party provide the plan timelines showing when facilities are to have their long-term actions completed?
See answer to question 70.

No. 77
Country: Canada
CNS-REF.-ART.: ARTICLE 19
PAGE OF REPORT: 166
CHAPTER OF NAT. 3.19.3.2.

In the third paragraph, the report states: “Such operational limits and conditions mainly arise from the Canadian experience on CANDU type reactor operation, transferred to CNE.” While the operational limits and condition (OLC) documents developed by Canadian utilities represent a logical starting point for developing OLCs for CNE, it is important to note that design differences across Canada’s CANDU fleet have necessitated slight differences in the OLCs used at each station. To what extent did Autoridad Regulatoria Nuclear (ARN) and/or Nucleoeléctrica Argentina S.A. (NA-SA) staff review the Canadian OLCs for applicability to CNE?
CNE together with Embalse Plant Designer (Candu Energy) have recently completed the analysis to determine the LOE (Limits of Operating Envelope) and SOE (Safe Operating Envelope).
The LOE determines the type of safety analysis applied to different plant parameters that intend to define the SOE. The SOE are the conservative values of a set of plant parameters that guarantee to accomplish the acceptance criteria in the safety analysis for the design basis events.
With the SOE, the Plant will set the OLC (Operating Limit Conditions). The Plant Operating Policies and Principles (OPP) Manual will be reviewed and will include in one annex the OLC; this will be finished and approved by ARN prior to the plant start up for second operation cycle.

No. 78
Country: Mexico
CNS-REF.-ART.: ARTICLE 19
PAGE OF REPORT: 178
CHAPTER OF NAT. Section 3.19.9.

What is the total amount of financial resources (U.S.Dlls.) for the final disposal of radioactive waste according with the provisions described at Act No. 25.018?
In order to secure the availability of sufficient resources, the current regulations set forth the creation of funds for financing the PNGRR (National Program for Radioactive Waste Management) and the decommissioning of each NPP. These funds shall come from the main radioactive waste generators, which are currently within the sphere of the State.
In accordance with the principle of unity of action and patrimony of the State, and while the nuclear power plants remain within the sphere of the State, the funding for PNGRR activities will depend on the National Budget granted to CNEA. Regarding long-term projects, such as the installation of future repositories, as long as the funds anticipated by the current regulation are not integrated, the national State shall secure the availability of sufficient resources for CNEA to deal, when necessary, with the expenditure and investments to finance the management of waste originated from nuclear power plants.

No. 79
Country: China
CNS-REF.-ART.: ARTICLE 19
PAGE OF REPORT: 18
CHAPTER OF NAT. Section 2.13.

It is mentioned that "Commissioning was planned to demonstrate that all components, systems and structures are functioning in accordance to previously calculated data and design documentation."
Question: According to the experience of commissioning, some components, such as explosion valve, are not demonstrated by commissioning activities. Could you please list items that are not demonstrated in commissioning, and try to analysis the usability of these items?
It is not really viable to list the items whose performance is not "demonstrated" during commissioning.
Perhaps the original wording is misleading: the components classified as "Class 1", including all the components of Safety Systems, are already qualified in terms of functional capability and robustness. In general this qualification is based on the use of standards, and only for components with innovative features the qualification is made by tests. Commissioning tests include the functional tests of safety systems, meant to verify that the systems performance is coherent with the "previously calculated data and design documentation". In other words, commissioning functional tests will cover thoroughly all the DBA scenarios, which are dealt by the operation of safety systems. DEC scenarios are dealt by systems that are functionally tested in a systematic way in all the cases that the system operating conditions are achievable. In general severe accidents scenarios are not tested during commissioning.

No. 80
Country: China
CNS-REF.-ART.: Article 19
PAGE OF REPORT: 166-167
CHAPTER OF NAT. Section 3.19.5.

The content of the CNA II's Operating Manual is more specific than the CNA I's .
Question: Why is not CAN I implemented the good practice?
CNA I and CNA II are facilities commissioned within a span of many years, hence the Operating Manuals of both facilities are not equally structured. Nonetheless, all the information covered by the Operating Manual of CNA II is also covered with the same amount of depth and detail in the Operating Manual of CNA I. The organization of the chapters and volumes is however different for both units, so for CNA I, the information is arranged in three parts, whereas for CNA II, the information is arranged in six parts. It is foreseen that the documentation of CNA I will be updated in the frame of the long term operation.

No. 81
Country: Romania
CNS-REF.-ART.: Article 19
PAGE OF REPORT: 166-167
CHAPTER OF NAT. Section 3.19.4.

How does the regulator review and inspect the verification and validation of emergency operating procedures and severe accident management guidelines?
The emergency operative procedures (EOP) as well as the SAMGs are mandatory documentation that the operator has to submit to the regulator. The EOPs are based on engineering judgment, experiences in other similar plants and validated against simulations in a full-scope simulator on site. As it is mentioned on page 33 of the national report “To develop and implement Severe Accident Management Guidelines (SAMG) specific for CNE, based on the Generic Guidelines developed by the CANDU Owner’s Group (COG), with the designer support. These SAMG will fulfill the Regulatory Body requirements related with the stress test that includes the consideration of the mitigation actions foreseen in order to prevent large radioactive releases as a consequence of damages to the core and the spent fuel pool”. The SAMG strategies are based on computer simulation with a validated system code for these purposes.

No. 82
Country: China
CNS-REF.-ART.: Article 19
PAGE OF REPORT: 168
CHAPTER OF NAT. Section 3.19.8.

It is mentioned in section 3.19 that "use of root cause analysis methodologies in the cases where an event is applicable in domestic plants". Normally, RCA method is applied to event investigation and analysis, especially in the internal event.
Question: How to apply the RCA method into analysis and assessment of the external event?
External events are events received as operating experience feedback and there is not root cause analysis for those events. They are used for lessons learned and possible corrective actions (Annex V).

No. 83
Country: China
CNS-REF.-ART.: Article 19.7.
PAGE OF REPORT: 163
CHAPTER OF NAT. 3.19.8.

Description in section 3.19.8 "Incidents significant to safety are reported in a timely manner by the holder of the relevant license to the ARN.”. "An event report is submitted to IRS when the event is considered by the national coordinator to be of international interest. During the period 2013-2015, three relevant events were reported to IRS."
Question: Are there any criteria to identify the Incidents significant to safety? Could you provide the criteria? How to make sure all the Incidents significant to safety are reported to the ARN correctly and completely. What are the criteria for the events reported to the IRS?
There are criteria to identify incidents significant to safety and they are contained in the Operation License from each NPP.
The criteria applied to identify the Incidents significant to safety are the ones expressed in the Argentine Regulation. These criteria are:
a. Safety Related System degradation: reactivity control, Heat Transport System pressure, flow or temperature; moderator system parameters; BOP parameters.
b. Safety System unavailability, safety system instrumentation and essential supplies.
c. Significant degradation of one of the radioactive safety barriers (fuel cladding, heat transport system and containment system).
d. Occupational exposure or radioactive effluents discharge to the environment, in excess of the corresponding authorized limits.
e. Internal or external events, both natural and resulting from human action, which could affect directly or indirectly the installation safety.
f. Any event considered by the facility Internal Safety Advisory Committee.

Each plant has an operating procedure where those criteria were expressed. Besides, station manager or shift supervisor could report to the regulator other events that at their own criteria are Incidents significant to safety.
For example:

a. Unplanned shutdown.
b. Damage to the equipment that could affect the safety, which were caused by inadequate operation, design flaws, quality material, maintenance and/or repair.
c. Heavy water spill greater than 100 kg.
d. Isotopic degradation is equal or greater than to 0.1% in the heat transport systems or moderator.

ARN makes sure that all the events are reported through regular inspections of NPP events data bases.
According with IRS rules, events reported to IRS must be useful as operating experience feedback to the nuclear community.

ANNEX III

MAIN TECHNICAL FEATURES OF THE ARGENTINE NUCLEAR POWER PLANTS IN OPERATION

III.1. ATUCHA I NUCLEAR POWER PLANT

III.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 9 km from Lima, Province of Buenos Aires, and near 100 km to the north-west of Buenos Aires city. ***Figure III.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 reactor is a pressure vessel type, fuelled with slightly enriched uranium and it is heavy water moderated and cooled (being of the PHWR type). The reactor 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 III.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 plant suffered two essential modifications that improved its performance:

- In 1977 bits 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.

III.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 III.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 plant service transformer.
13. Generator transformer.
14. Off-site system transformer.
15. Secondary Heat Sink.
16. Emergency Power Supply.

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 III.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 Secondary Heat Sink (SHS) building is located adjacent and north of the reactor building. It consists of two identical trains, each with instrumentation, control equipment and power separate and independent. Its main functions are: feeding on the secondary side of both SGs with demineralized cold water (due to the unavailability of the SGs and the startup and shutdown system), and removal of both the decay heat from fuel elements and the heat stored in the PHTS by evaporation and release to atmosphere via the steam plant vent.

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 high-voltage plant 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.

The Emergency Power Supply building (EPS) is located in front of the Switchgear building. The structure of the building was designed according to the quality standards requirements for the nuclear industry considering earthquake, tornadoes and fire. The Emergency Power Supply building (EPS) houses three emergency diesel generators with 100% capacity each one of them, its instrumentation and control equipment, the secured bus bars (BU/BV), the non-interruptible bus bars (EM/EN), batteries and transformers. The Emergency Power Supply system consists of two identical independent electric trains physically separated.

III.1.3. CNA I Main Systems

In what follows the main safety and process systems that are part of the plant, are briefly summarised.

III.1.3.1. Reactor

The reactor ***(Figure III.1-5)*** is of the pressure vessel type, 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 253 tubes arranged vertically in a trigonal lattice within the moderator tank, of which 250 tubes are cooling channels and contain slightly enriched uranium fuel assemblies, two tubes have been instrumented to measure level and the other tube is empty and is used for inspection of the moderator tank internals during programmed outages. The fuel assemblies are bundles of 37 closely packed fuel rods which are arranged in 4 concentric rings having 1, 6, 12 and 18 fuel rods each. 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. 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.

III.1.3.2. Reactor coolant system and moderator system

The reactor coolant system ***(Figure III.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.

III.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 tilter 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 tilter 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 fuel transfer tube connects the reactor building to the transfer pool in the fuel pool building.

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.

III.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.

III.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.

III.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.

III.1.3.7. Electric power system

The Atucha I nuclear power plant has two physically independent grid connections ***(Figure III.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 plant auxiliary service requirements by means of one high-voltage plant service transformer.

The high-voltage plant 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 plant 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. In this case, an power source consisting of D.C./A.C. converters ensures continuous supply to the uninterruptable loads. For this purpose the appropriate converters are provided.

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 plant 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 plant service transformer as well as the water turbine driven generator (located in the water turbine building).

The 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 BU and BV buses-, which are usually supplied by 6.6 kV buses BA and BB.

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 case of failure of the auxiliary power system, the emergency power for the emergency power system is provided by two diesel generator set. The capacity of the set is designed such that the emergency power required for power plant shutdown and mitigation of the design-basis accidents can be supplied by one of the three diesel generator sets (3 x 100%). In this situation, only safety related loads are fed. Each emergency diesel generator is, in turn, constituted by different main and auxiliary subsystems, such as the starting subsystem, the lubrication subsystem, etc.

III.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 second heat sink system (SHS) is a back- up system aimed to remove the core decay heat during fault conditions with unavailability or ineffectiveness of regular heat sinks (including events related to SBO), by using light demineralizing water cooling. The SHS is completely independent of existing systems at the plant which provides the SGs' feeding and venting. The SHS 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 two DGs that are mechanically and directly coupled to each electrical generator, and through reduction gears to each pump as well as to I&C.

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.

III.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 fuel assemblies	250
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 plant service transformer rated power	35 / 20 / 20 MVA
High-voltage plant 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

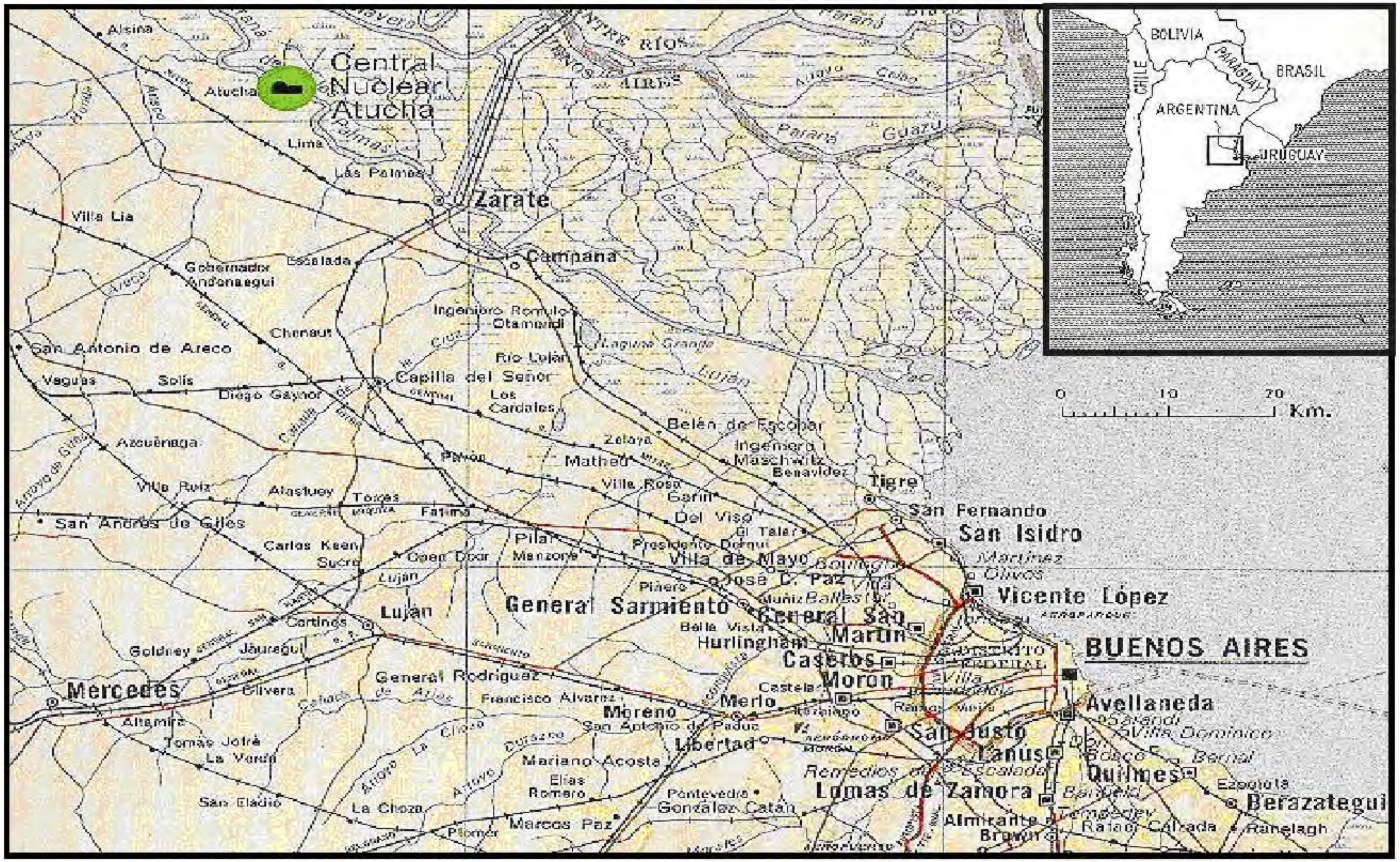


Figure III.1-1 - Atucha I Nuclear Power Plant - Geographic Location

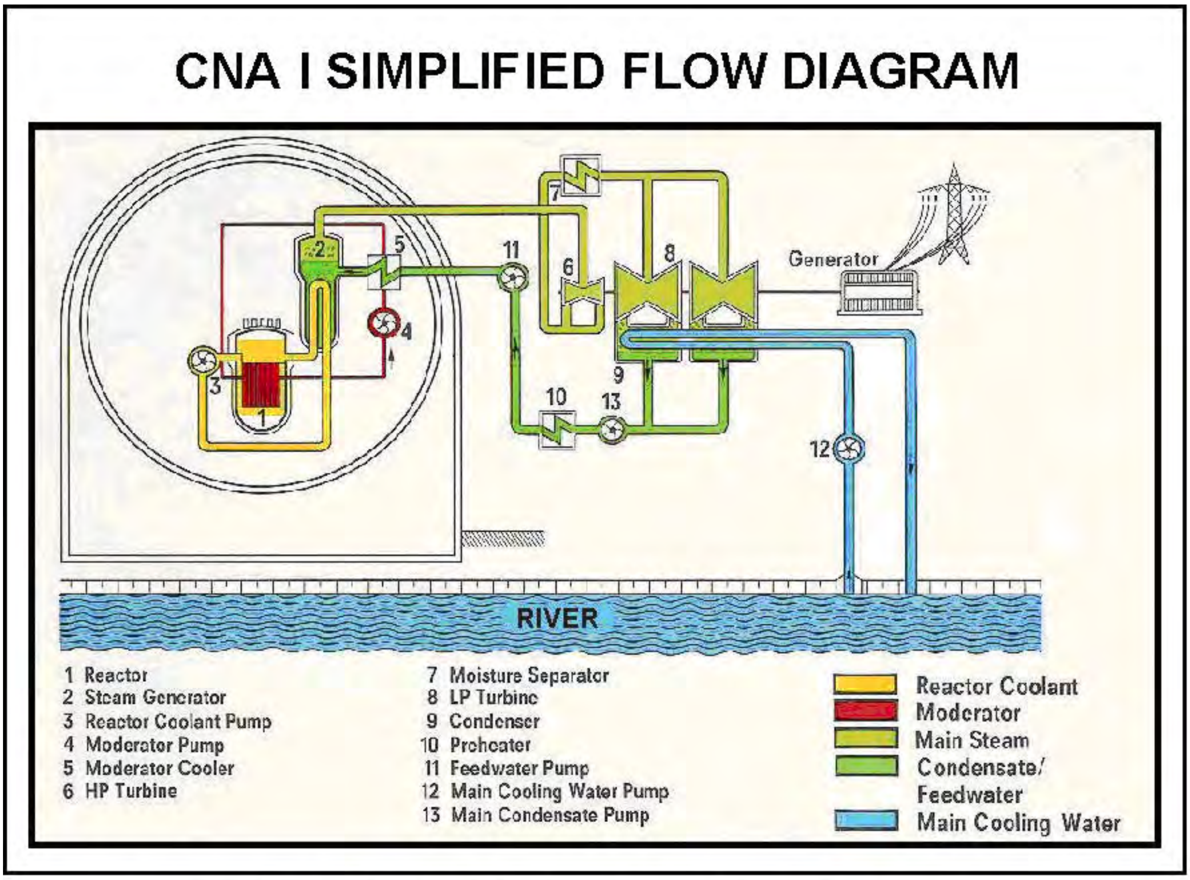


Figure III.1-2 - Atucha I Nuclear Power Plant - Main Systems

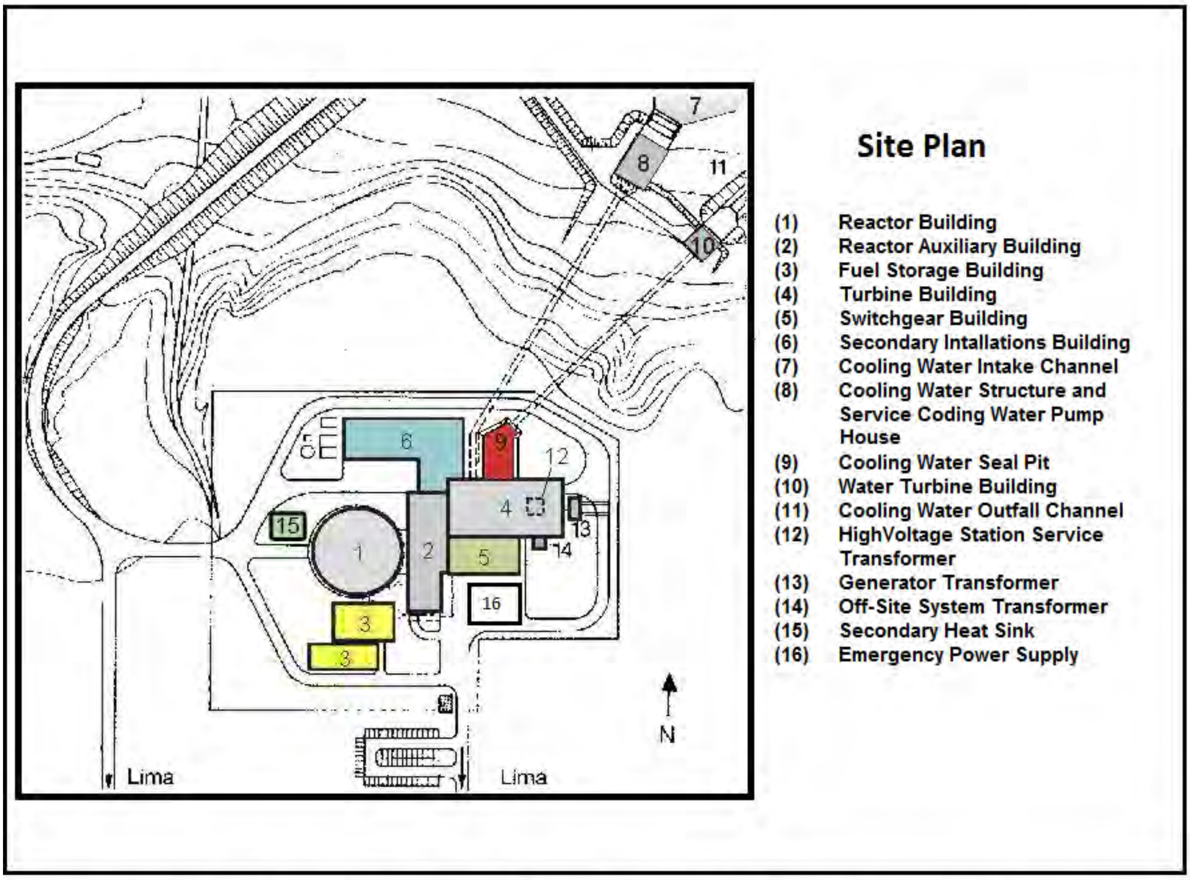
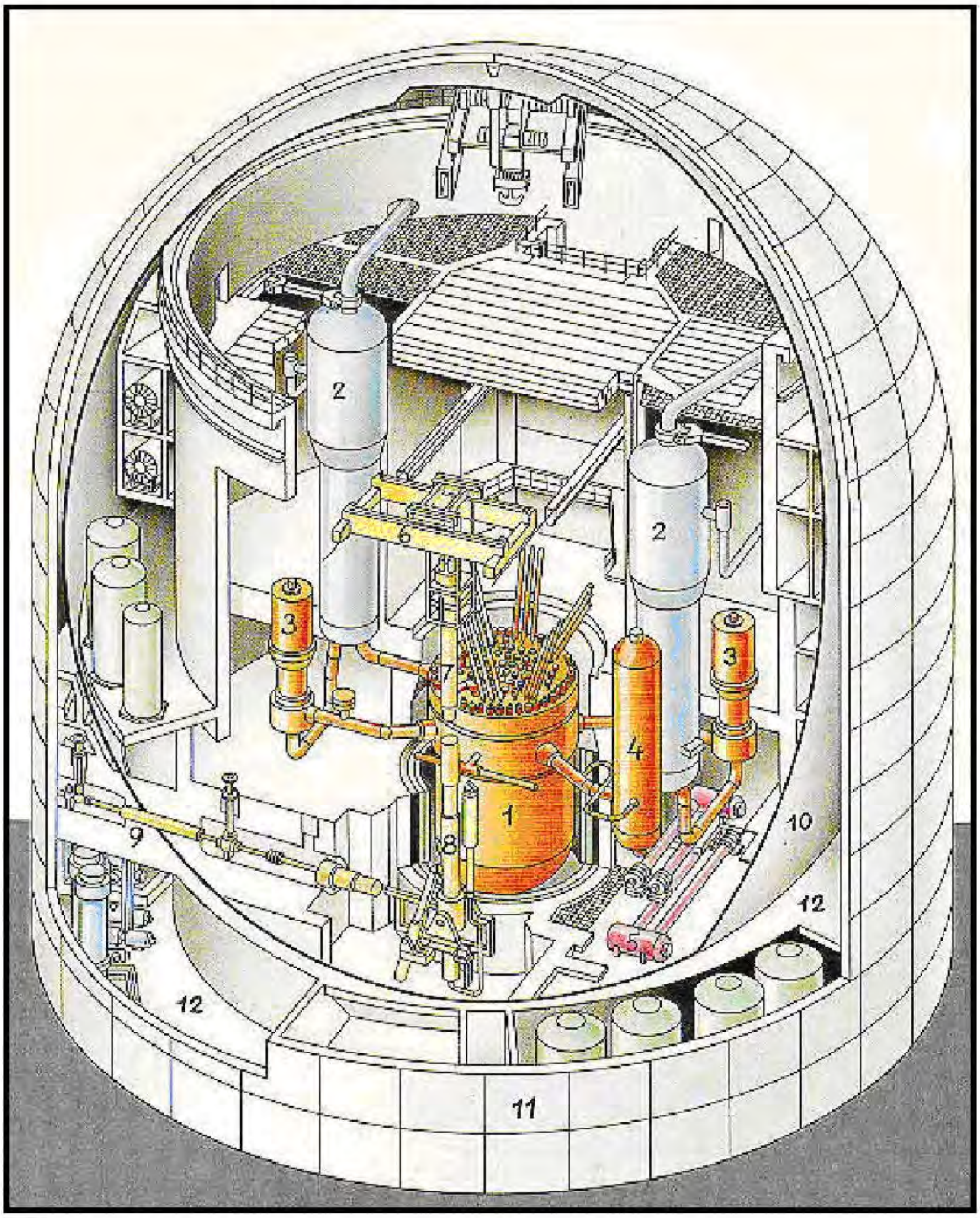


Figure III.1-3 - Atucha I Nuclear Power Plant - Main Building and Structures

REACTOR BUILDING



1 -	Reactor pressure vessel	7 -	Refueling machine
2 -	Steam generator	8 -	Tilter
3 -	Reactor coolant pump	9 -	Fuel transfer tube
4 -	Pressurizer	10 -	Containment
5 -	Moderator cooler	11 -	Reactor Building.
6 -	Refueling machine travelling gear	12 -	Annulus

Figure III.1-4 - Atucha I Nuclear Power Plant - Reactor Building

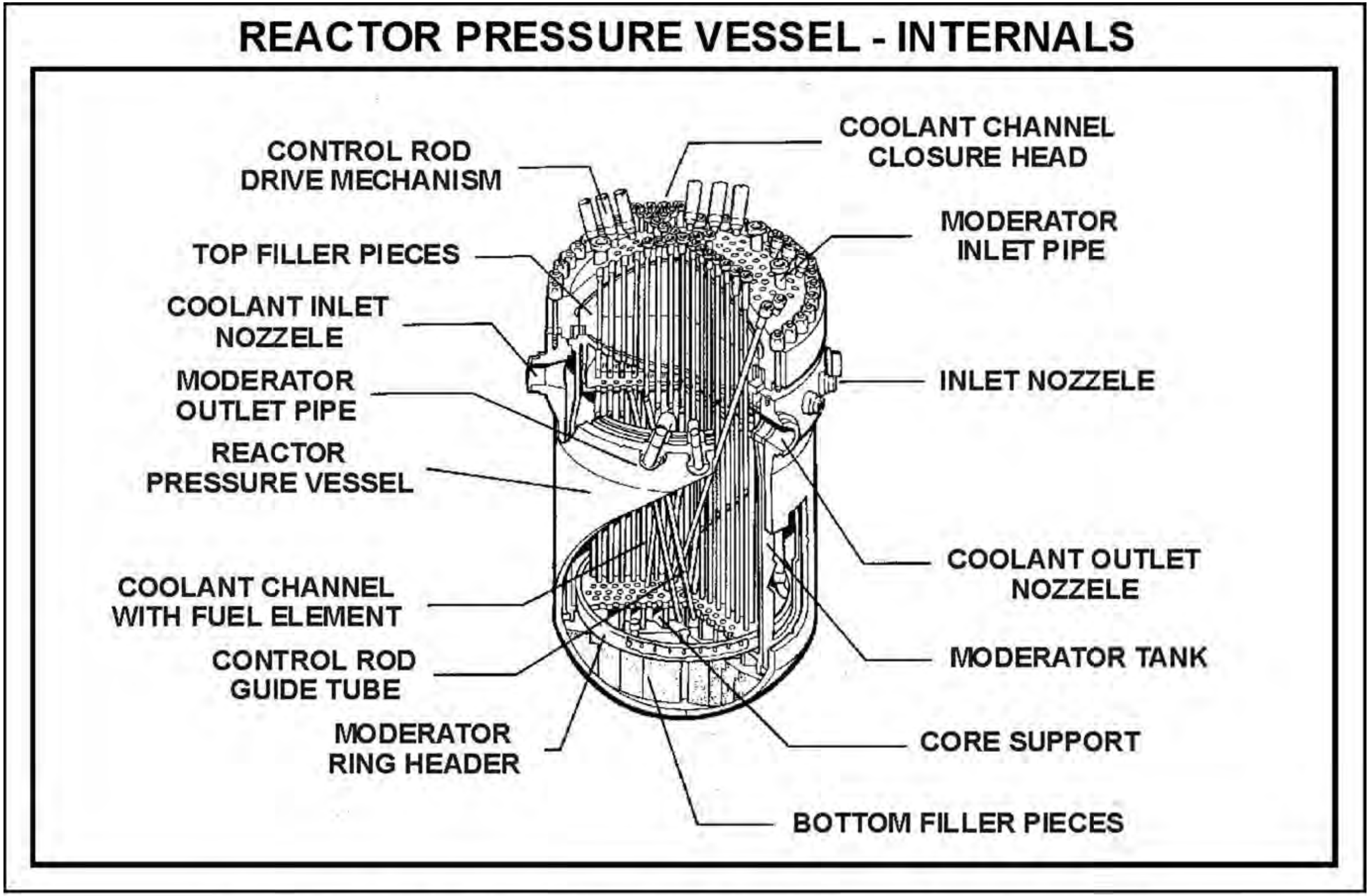
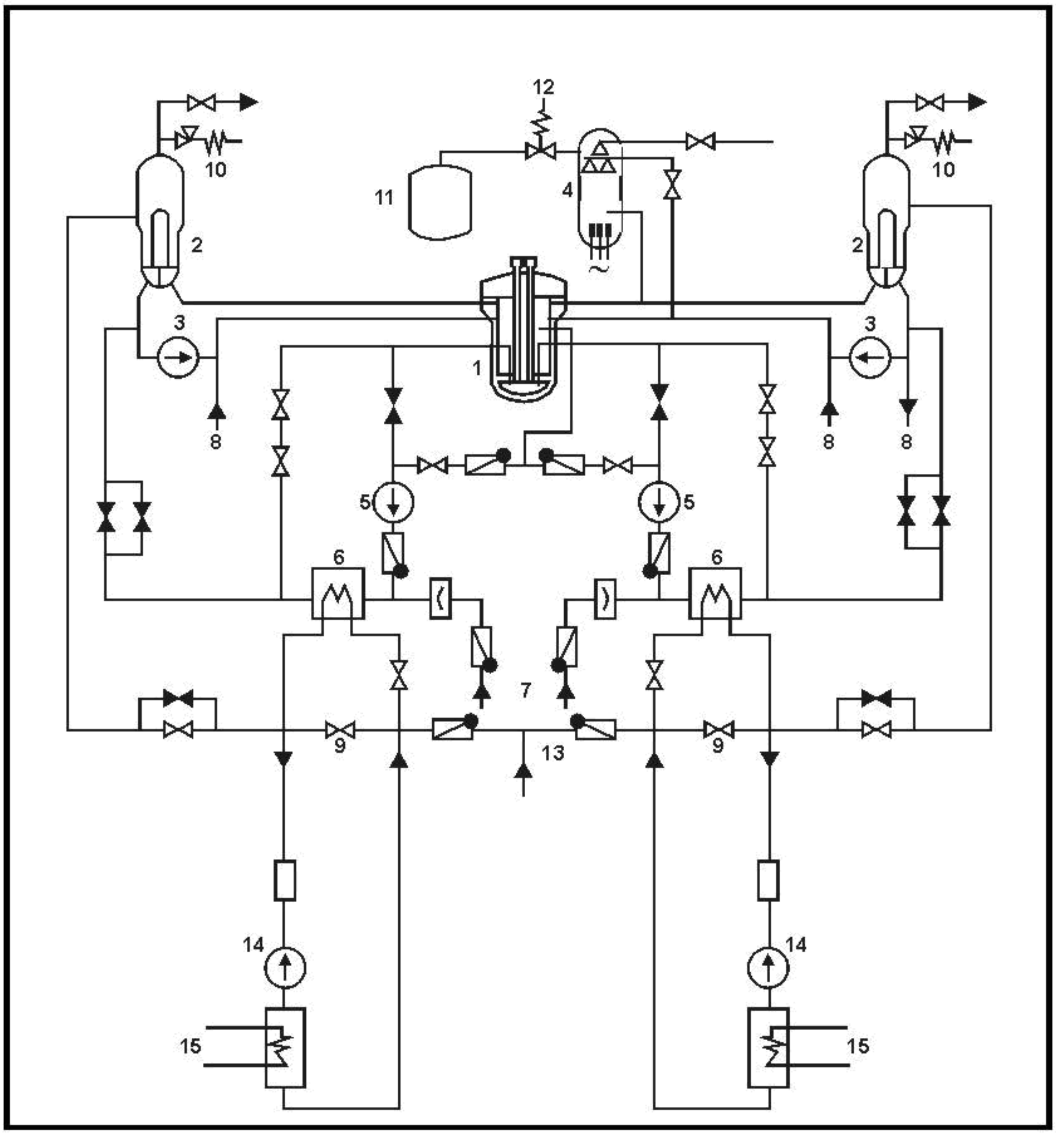


Figure III.1-5 - Atucha I Nuclear Power Plant - Reactor Pressure Vessel and Internals

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 III.1-6 - Atucha I Nuclear Power Plant
Reactor Coolant System and Moderator System

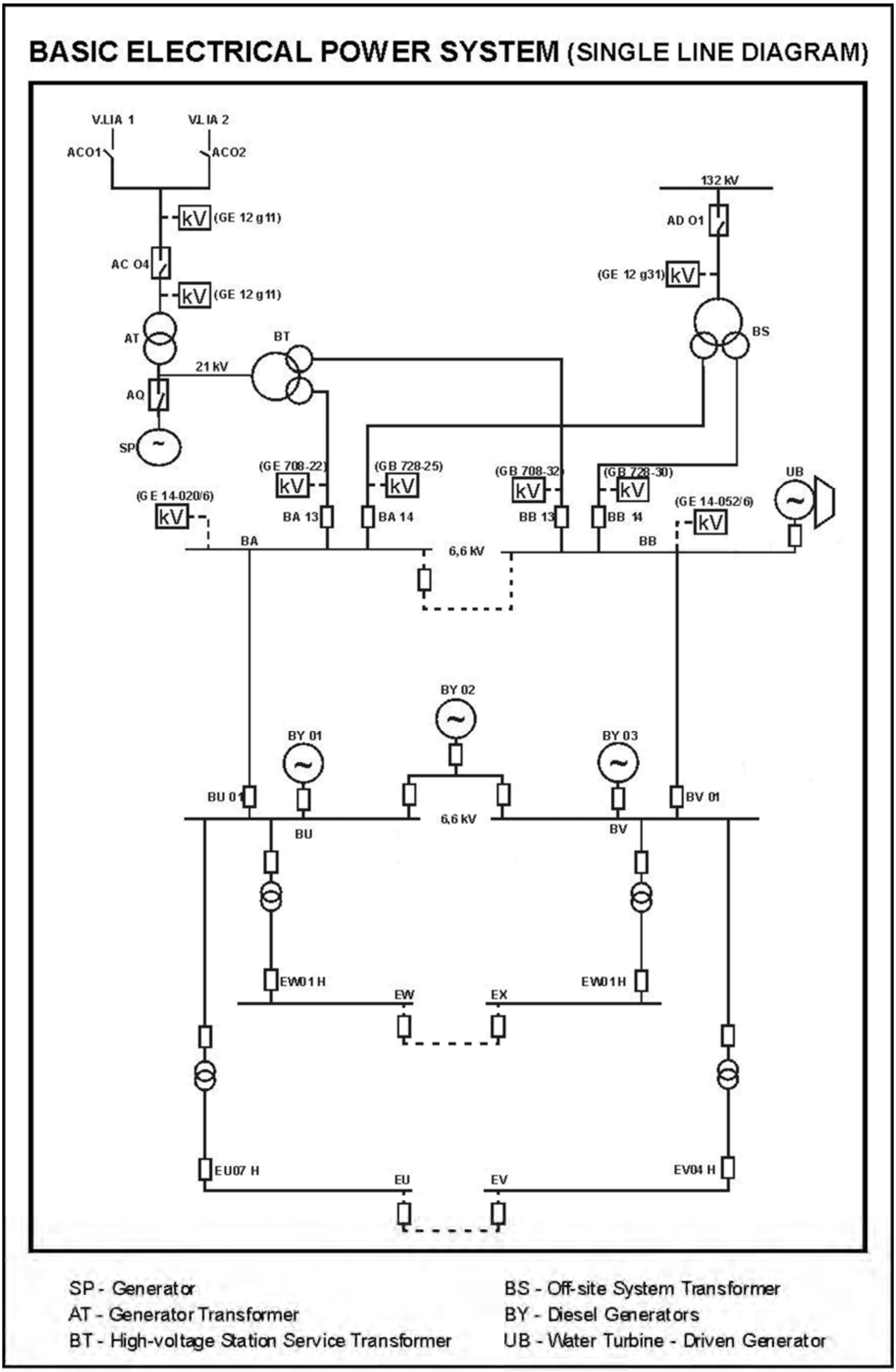


Figure III.1-7 - Atucha I Nuclear Power Plant
Basic Electrical Power System

III.2. EMBALSE NUCLEAR POWER PLANT

III.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 III.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 III.2-2 shows schematically the main systems of the Embalse Nuclear Power Plant.

III.2.2. Overall Plant Layout

Building and structure arrangements of CNE are shown in *Figure III.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 III.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.

III.2.3. CNE Main Systems

In what follows the main safety and process systems that are part of the station, are briefly summarized.

III.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 III.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 rod. 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 rods are welded to two zirconium alloy end plates to form the cylindrical bundle. The end plates maintain separation among the fuel rods at the bundle element extremities.

The separation among the fuel rods at the bundle mid-length is maintained by the spacers which are brazed to the fuel rods. The spacers are positioned on each individual fuel rod such that the contact between any two mating rods is spacer-to-spacer. Bearing pads are brazed to the outer ring of fuel rods. 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 the shutoff units and the liquid poison injection system; these systems are discussed in section III.2.3.9.

III.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 III.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.

III.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 III.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.

III.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 Iifting 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 spent fuel assemblies are introduced in stainless steel baskets, each of them containing up to 60 spent fuel assemblies 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.

III.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.

III.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.

III.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 systems 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.

III.2.3.8. Electric power system

The Embalse Nuclear Power Plant has two physical independent grid connections (Figure III.2-8). One of them is the 500 kV grid and the other is the 132 kV 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.

III.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 III.2.3.8.

III.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	*637,5* MWe
Bulk nominal electric power	*682,5* MWe
Authorized thermal power	*2064* 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

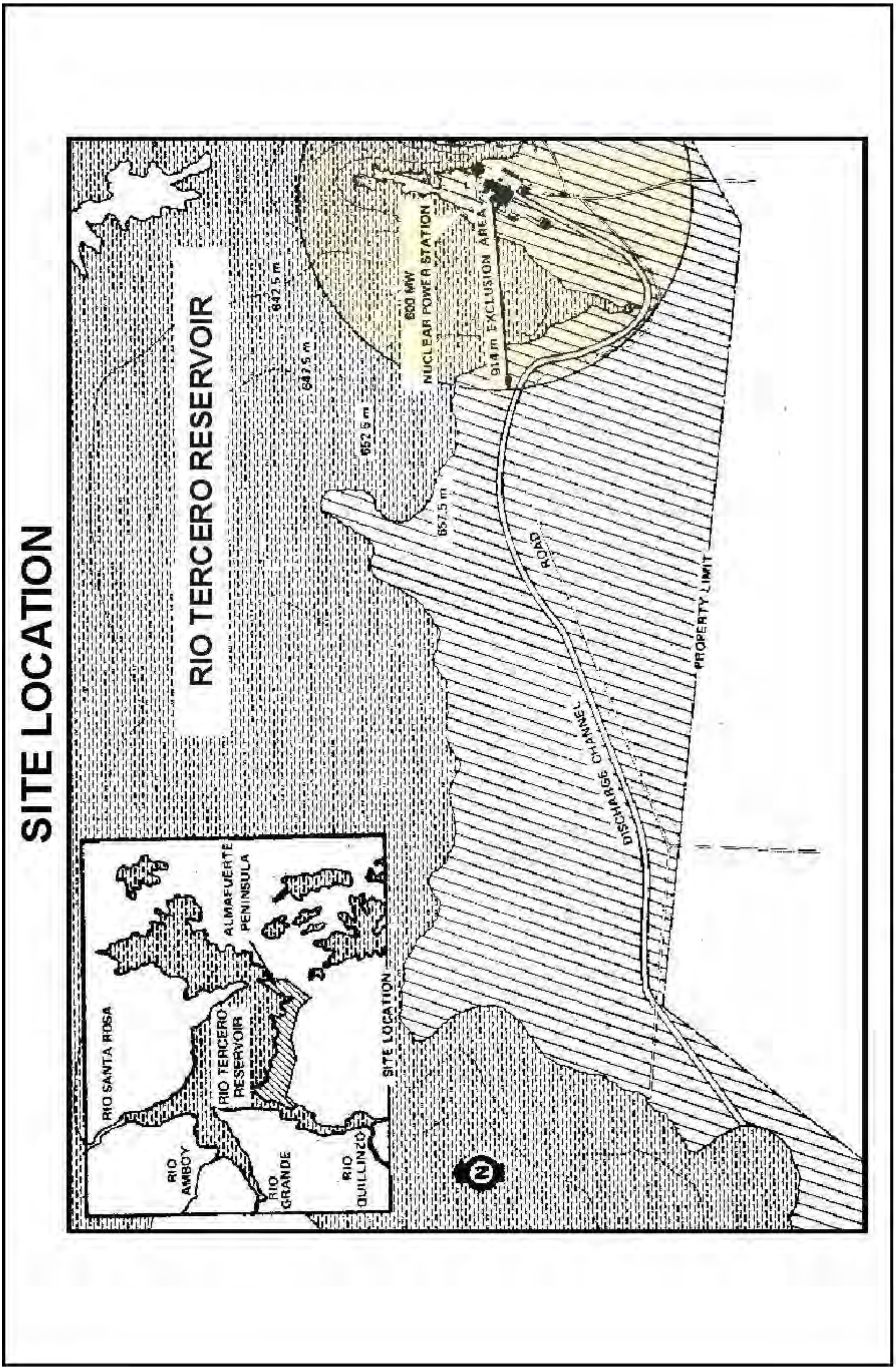


Figure III.2-1 - Embalse Nuclear Power Plant - Site Location

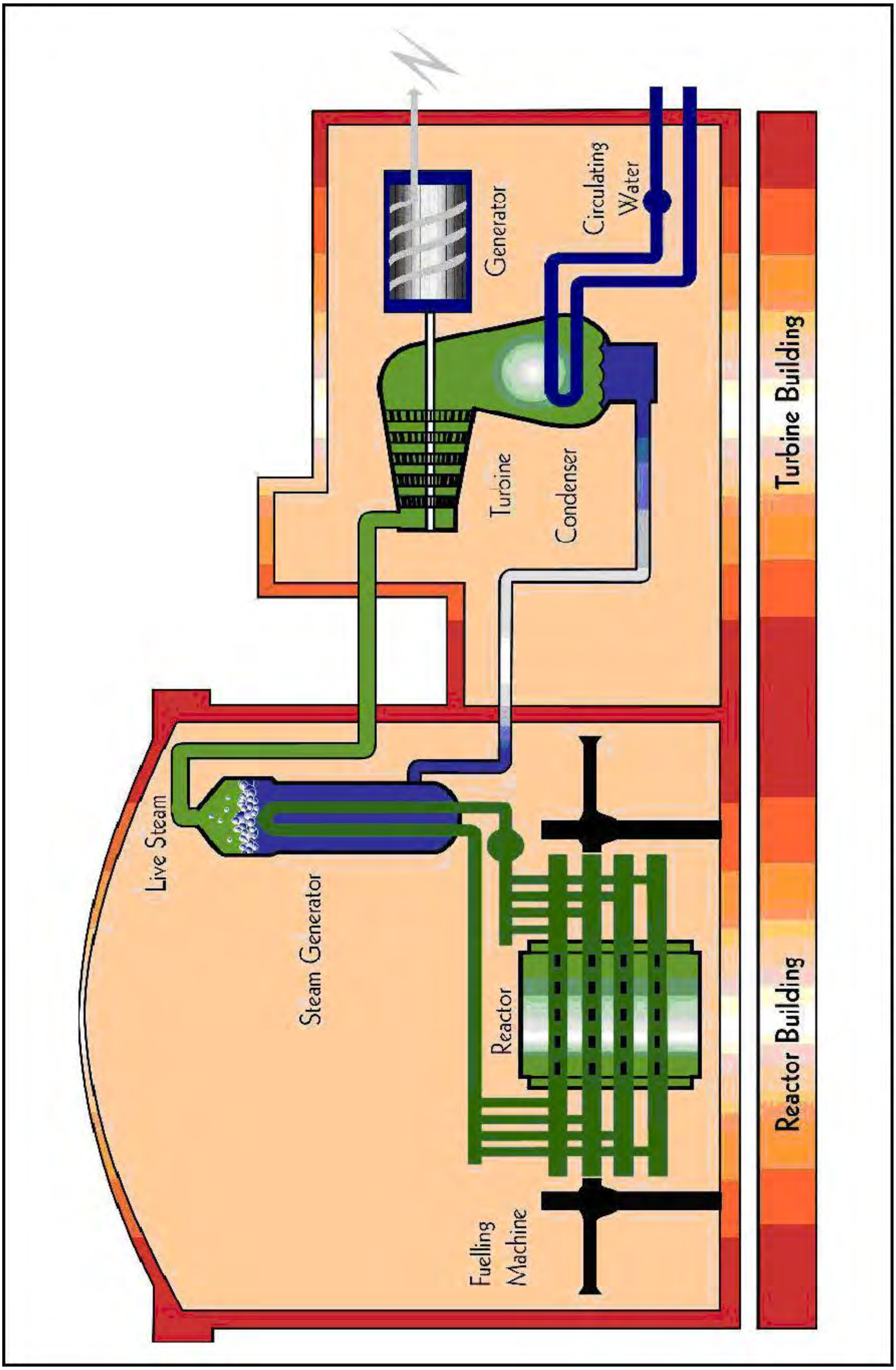
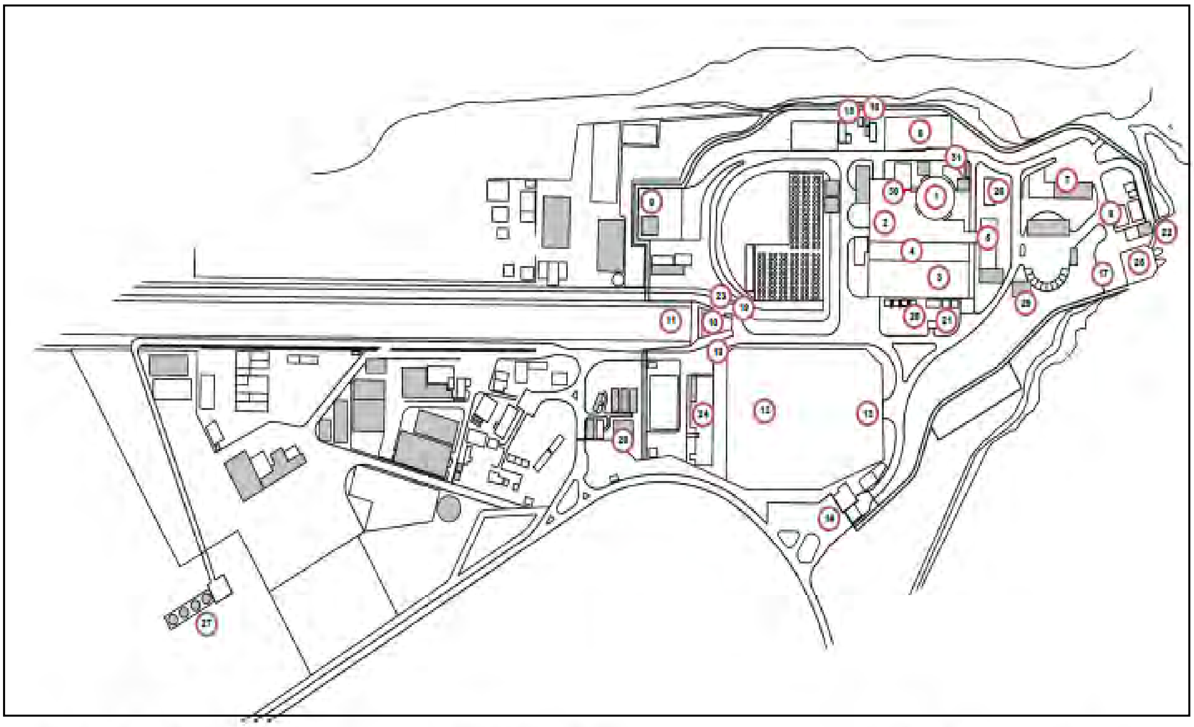


Figure III.2-2 - Embalse Nuclear Power Plant - Simplified Flow Diagram

SITE PLAN



1- Reactor Building
2- Service Building
3- Turbine Building
4- Auxiliary Bay
5- Administration Building
6- Water Treatment Plant
7- Garage
8- Class III Diesel 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- Firefighting Pumps
19- Drain Pumps
20- Transformer Area
21- Hydrogen Storage
22- Emergency Water System Pump House
23- Process Water Pool
24- General Warehouse
25- Pump House
26- High Pressure Emergency Core Cooling Building
27- Retubing waste silos for Life Extension – CANISTER
28- Control Room Simulator Building
29- Emergency Power Supply Building
30- Emergency Ventilation System Building
31- Retubing Operation Control Room for Life Extension

Permanent Buildings

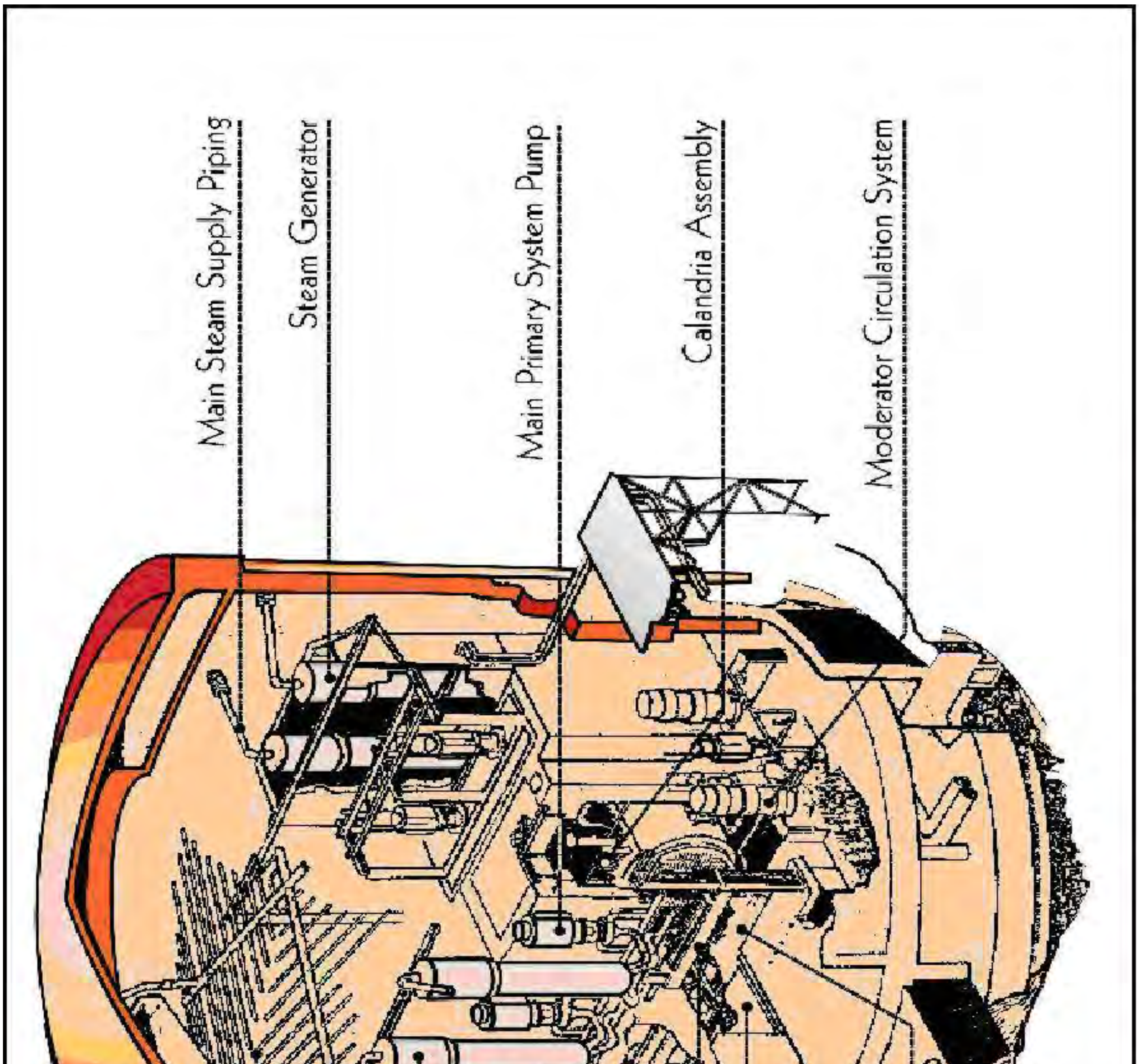
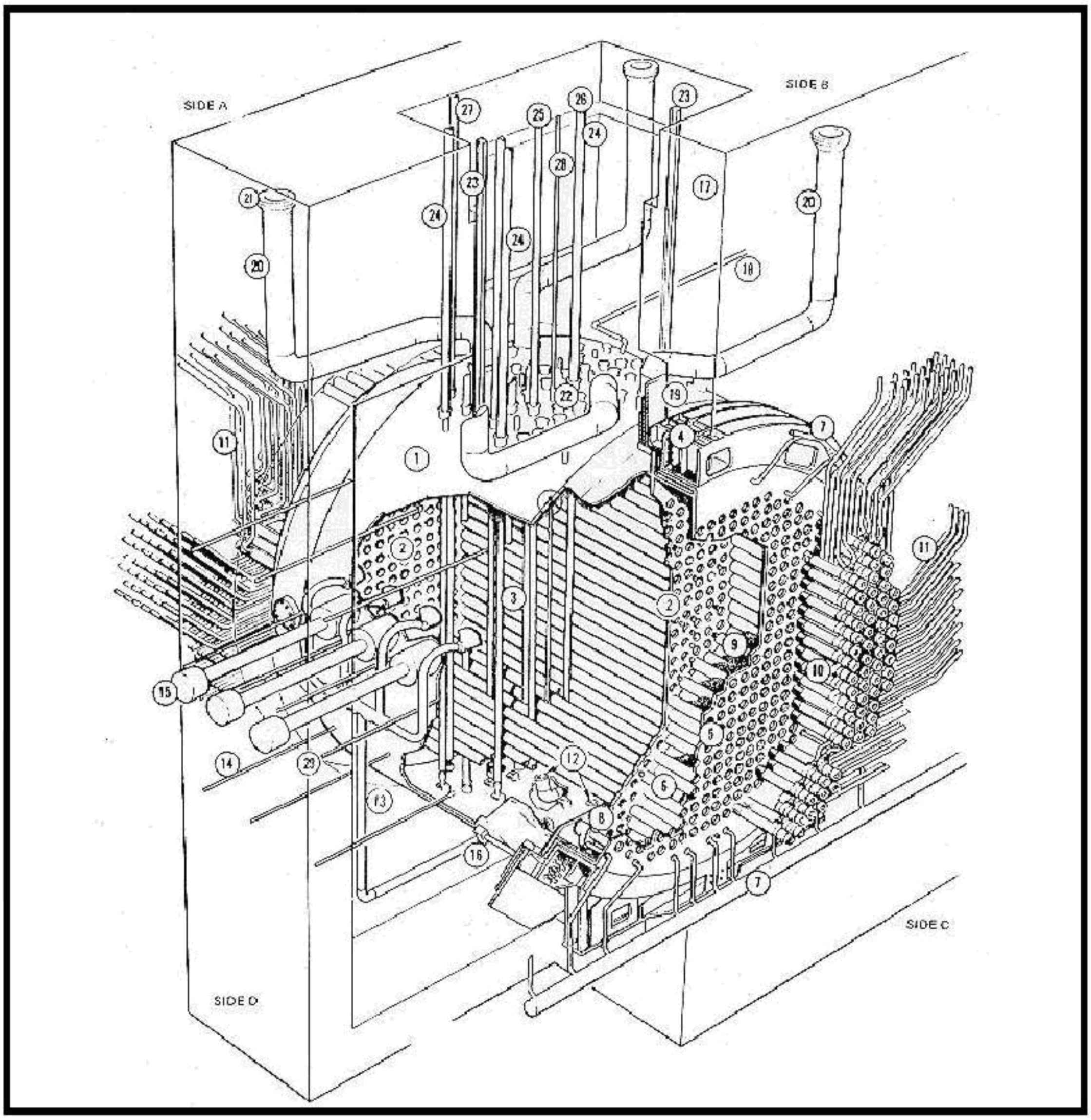


Figure III.2-4 - Embalse Nuclear Power Plant - Reactor Building

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 III.2-5 - Embalse Nuclear Power Plant - Reactor Assembly

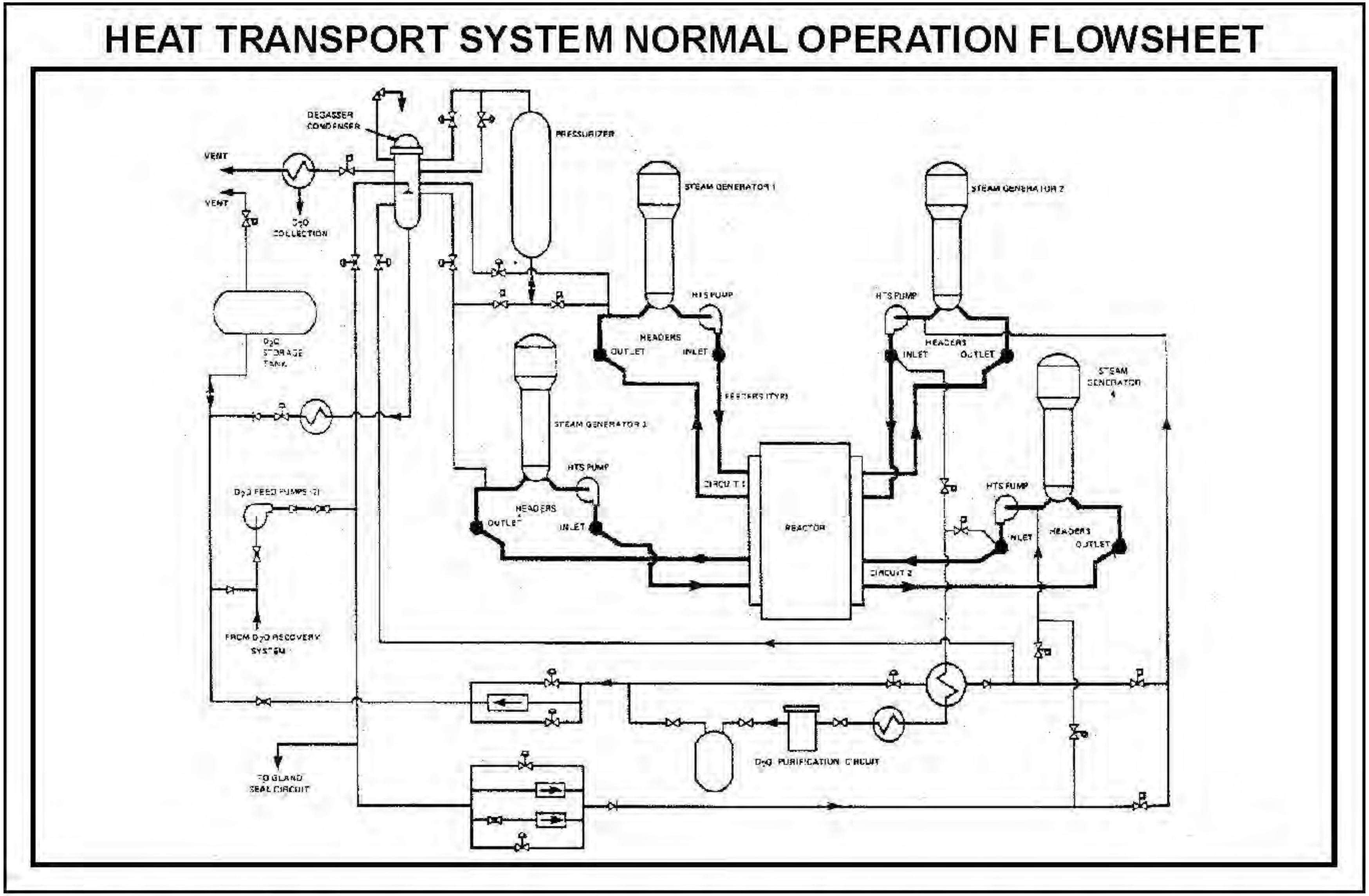


Figure III.2-6 - Embalse Nuclear Power Plant - Heat Transport System Normal Operation Flowsheet

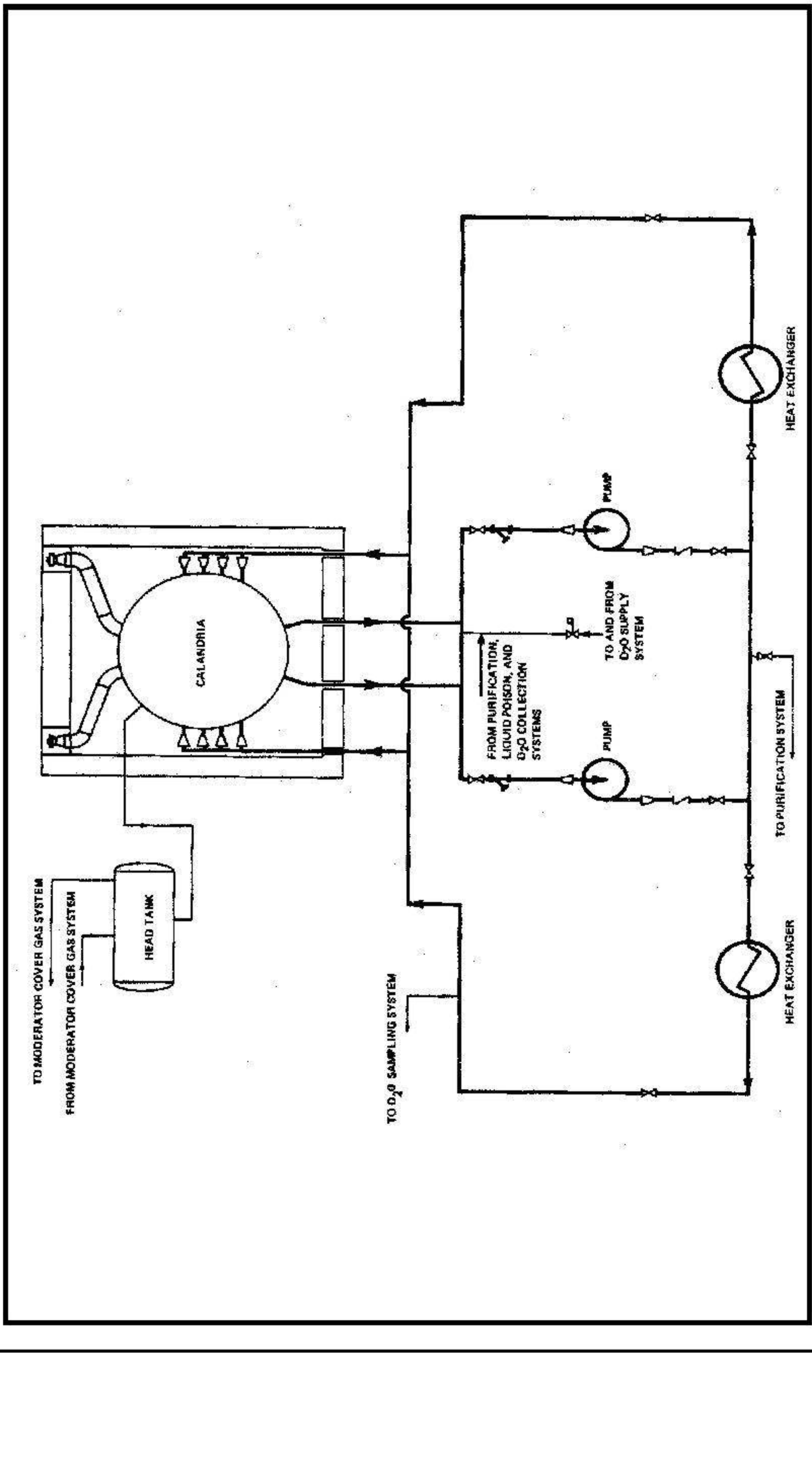


Figure III.2-7 - Embalse Nuclear Power Plant - Simplified Moderator System Flow Diagram

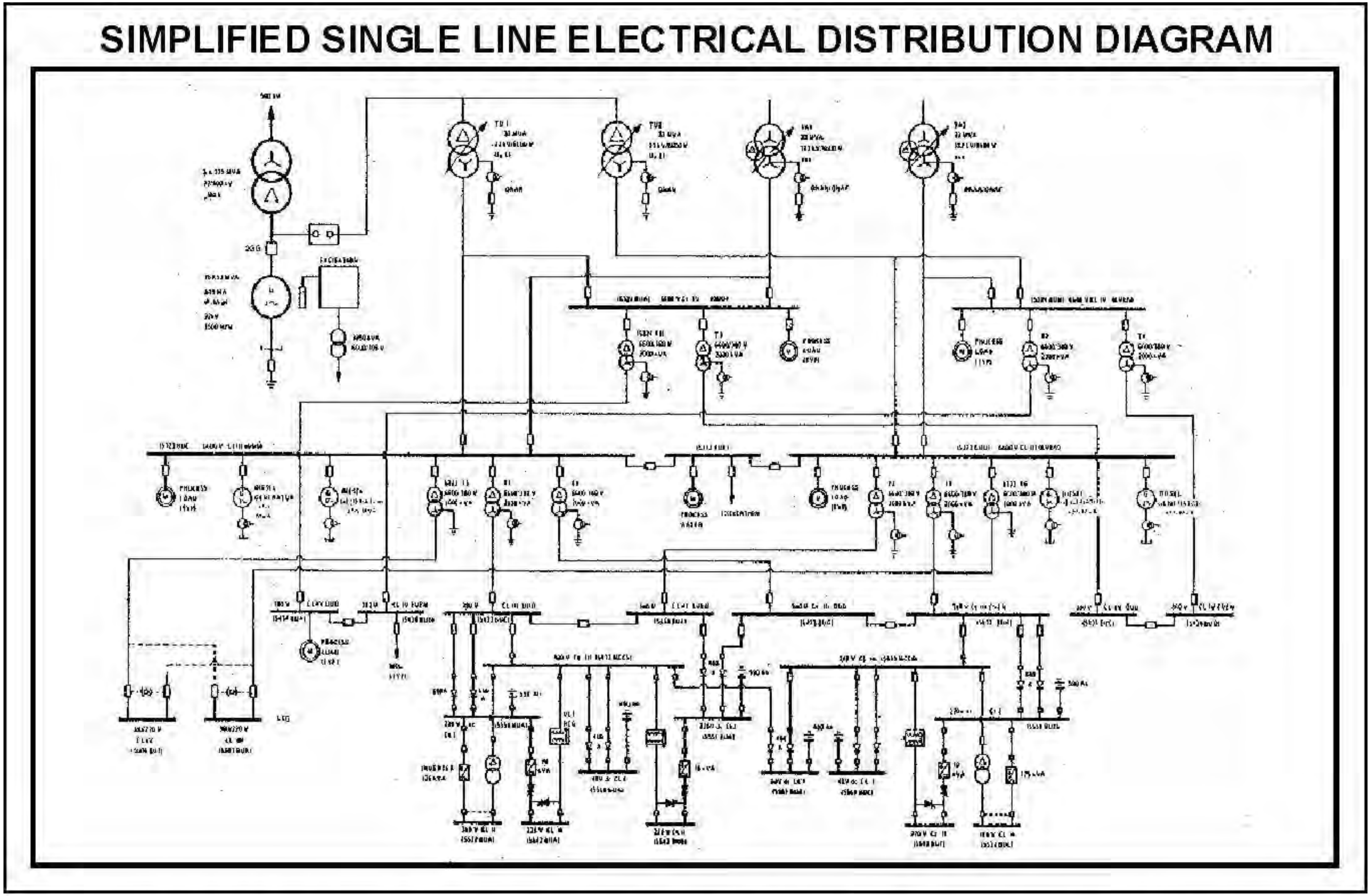


Figure III.2-8 - Embalse Nuclear Power Plant - Simplified Single Line Electrical Distribution Diagram

ANNEX IV

PRINCIPAL TECHNICAL CHARACTERISTICS OF CAREM REACTOR PROTOTYPE

IV.1. INTRODUCTION

CAREM reactor prototype (CAREM 25) has an indirect cycle reactor with some distinctive and characteristic features that greatly simplify the design, and also contributes to a higher safety level. Some of the high level design characteristics of the plant are: integrated primary cooling system, natural circulation, self-pressurised primary system and safety systems relying on passive features.

IV.2. PRIMARY CIRCUIT AND ITS MAIN CHARACTERISTICS

CAREM 25 NPP design is based on a light water integrated reactor. The whole high-energy primary system, core, steam generators, primary coolant and steam dome, is contained inside a single pressure vessel ***(Figure IV-1).***

The flow rate in the reactor primary systems is achieved by natural circulation. Figure IV-1 shows a diagram of the natural circulation of the coolant in the primary system. Water enters the core from the lower plenum. After having heated the coolant exits the core and flows up through the riser to the upper dome. In the upper part, water leaves the riser through lateral windows to the external region. Then it flows down through modular steam generators, decreasing its enthalpy. Finally, the coolant exits the steam generators and flows down through the down-comer to the lower plenum, closing the circuit. The driving forces obtained by the differences in the density along the circuit are balanced by the friction and form losses, producing the adequate flow rate in the core in order to have the sufficient thermal margin to critical phenomena. Reactor coolant natural circulation is produced by the location of the steam generators above the core. Coolant acts also as neutron moderator.

Self-pressurization of the primary system in the steam dome is the result of the liquid-vapor equilibrium. The large volume of the integral pressuriser also contributes to the damping of eventual pressure perturbations. Due to self-pressurisation, bulk temperature at core outlet corresponds to saturation temperature at primary pressure. Heaters and sprinkles typical of conventional PWRs are thus eliminated.

IV.3. REACTOR CORE AND FUEL DESIGN

The core has Fuel Assemblies (FA) of hexagonal cross section. Each fuel assembly contains 108 fuel rods of 9 mm outer diameter, 18 guide thimbles and 1 instrumentation thimble ***(Figure IV-2)***. Its components are typical of the PWR fuel assemblies. The fuel is enriched UO_2. Core reactivity is controlled by the use of Gd_2O_3 as burnable poison in specific six fuel rods and movable absorbing elements belonging to the Adjust and Control System. Chemical compounds are not used for reactivity control during normal operation. The fuel cycle can be tailored to customer requirements, with a reference design for the prototype of 420 full-power days and 50% of core replacement.

Each Absorbing Element (AE) consists of a cluster of rods linked by a structural element (namely “spider”), so the whole cluster moves as a single unit. Absorber rods fit into the guide tubes. The absorbent material is the commonly used Ag-In-Cd alloy. AE are used for reactivity control during normal operation (adjust and control system), and they are also used to produce a sudden interruption of the nuclear chain reaction when required (fast shutdown system).

IV.4. STEAM GENERATORS

Twelve identical 'Mini-helical' vertical steam generators, which are of the "once-through" type are placed equally distant from each other along the inner surface of the RPV ***(Figure IV-3)***. They are used to transfer heat from the primary to the secondary circuit, producing superheated dry steam at 47 bar. The secondary system circulates upwards within the tubes, while the primary goes in counter-current flow. The lay-out guarantees that the primary system flows through the steam generators. In order to achieve a rather uniform pressure-loss and superheating on the secondary side, the length of all tubes is equalized. Due to safety reasons, steam generators are designed to withstand the primary pressure without pressure in the secondary side and the whole live steam system is designed to withstand primary pressure up to isolation valves (including the steam outlet / water inlet headers) in case of SG tube breakage.

IV.5. REACTOR AUXILIARY SYSTEMS

Figure IV-4 shows a diagram of the main reactor auxiliary systems chemical / volume control system.

IV.5.1. Chemical / volume control system

This system maintains a high degree of water purity within the RPV and allows controlling the water level while volume changes are produced by the operating conditions. The water removed from the RPV is cooled in a letdown heat-exchanger, undergoes a stage of pressure reduction, is treated in filters, resin beds, de-gasifier columns and returned to the RPV by the charging pumps through a regenerative heat exchanger. A control volume tank provides a volume reservoir that allows to contain all the water surplus of the RPV from the operation in a solid way at 45 °C until full power operation.

IV.5.2. Suppression pool cooling and purification system

This system cools and purifies the suppression pool and the cooling pool for the residual heat removal system. The cooling system has redundancies: each branch has a plate heat exchanger and a pump, while both share the ion exchange bed for water purification. In the event of a LOCA, this system is capable of feeding pure water into the RPV.

IV.5.3. Shutdown cooling system

This system has two functions:

- To cool RPV water, removing decay heat during standard shutdown and refueling.
- To heat RPV water during plant start-up by an auxiliary steam system.

It is also redundant, since each branch comprises a pump, shell and tube heat exchanger for heating and for cooling.

IV.5.4. Components cooling system - closed external circuit

The components cooling system supplies cooling water to the systems that may contain radioactivity, providing a barrier among the radioactive fluid and the closed external circuit. It is redundant and comprises pumps and heat exchangers.

The closed external circuit is also redundant. It has cooling towers, pumps and supply tanks.

IV.5.5. Fuel pool cooling and purification system

It removes the heat resulting from nuclear decay of stored fuel elements and purifies pool water. The cooling system comprises two circuits - one in stand-by - each with a heat exchanger and a pump. In addition to a pump for the resin beds, the ion exchange bed, with its pre filters and post filters for water purification, as well as the skimming circuit are integrated by a pump, a tank and a filter.

IV.5.6. Control rod drive - hydraulic system

This system circulates water from the RPV to operate and maintain the Control Rods in position. It has two pumps in operation, to enhance system availability, as well as filters, valves for step-wise Control Rod motion and for operating Safety Rods rising, and redundant SCRAM valves.

IV.6. OPERATING CHARACTERISTICS

The natural circulation of coolant produces different flow rates in the primary system according to the power generated (and removed). Under different power transients a self-correcting response in the flow rate is obtained.

Due to the self-pressurizing of the RPV (steam dome) the system keeps the pressure very close to the saturation pressure. At all the operating conditions this has proved to be sufficient in order to guarantee a remarkable stability of the RPV pressure response. The control system is capable of keeping the reactor pressure practically at the operating set point through different transients, even in case of power ramps. The negative reactivity feedback coefficients and the large water inventory of the primary circuit combined with the self-pressurization features make this behaviour possible with minimum control rod motion. It concludes that the reactor has an excellent behaviour under operational transients.

IV.7. TURBINE GENERATOR PLANT SYSTEMS

The CAREM has a standard steam cycle of simple design. The twelve steam generators are connected alternately in two groups of six to an annular collector. Each branch has its own relief and isolation valves and finally they are joined to deliver the steam to the turbine. A single turbine is used.

CAREM secondary circuit is not a safety-graded system; the nuclear safety of the plant does not rely on the functioning of the steam circuit.

IV.8. INSTRUMENTATION AND CONTROL SYSTEMS

The control and supervision system is a “real time” computerized system for the control and supervision of the plant operation. This system includes the control centers, the information processing centers, the man-machine interfaces, the automatic systems for process control, sensors, actuators and a communication net that connect these systems. The general architecture of the system has four hierarchical levels clearly identified for the processes and three communication levels. The process levels are:

- Supervision level: composed by a net of supervision nodes. At this level occur all the man-machine interactions between the operators and the supervision system.
- Information level: composed by a net of information nodes.
- Control level: Composed by a net of control units. These units constitute the connection with the field units.
- Field level: Composed by a net of field units that are the connection with the sensors and actuators and by all the sensors and actuators of the control and supervision system.

IV.9. REACTOR PROTECTION SYSTEM

The design of the reactor protection system was performed according to the most advanced technology for nuclear power plants design, the “defence in depth” principle and the early failure detection, with the objective of avoiding the occurrence of accidents beyond the design base.

The reactor protection system has two independent subsystems. The first subsystem, is responsible for the generation of the first shutdown system trip signal, consists in a combination of hard logic and digital processing modules. The second subsystem, is responsible of the generation of the second shutdown system trip signal, and is achieved through diversity principle for the first and second shutdown systems.

The reactor protection system has four independent and redundant channels with voting and protective logic of dynamic type. This allows a high availability and reliability.

The main applied design criteria are:

- Physical and electrical independence.
- Functional diversity.
- Reduced size and robustness.
- Tolerance of failure.
- Possible in operation testing.
- Safe failure.

The interaction between the protection system and the control system is performed through electrical isolation. The interfaces are designed in order to avoid that any protection action could be inhibited by a control system action. The design guarantees that once a protective action is initiated it will be completed.

IV.10. ELECTRICAL SYSTEMS

The electrical loads are divided in three classes:

- Class I: DC, no supply interruption is admitted.
- Class II: AC, no supply interruption is admited.
- Class III: AC, supply interruption is admitted during a certain period.
- Class IV: AC, Supply interruption is admitted.

Classes I, II and III correspond to the Safety-related system. Class IV includes all the conventional systems. The electrical power supply corresponding to class I, II and III systems are distributed by two systems of independent buses. This redundant system is separately connected to each bus with independent layout and connections. Both bus systems can be interconnected in case of failure.

Auxiliary generators will supply power to the essential systems in case of no power generation or external supply. These auxiliary generators are redundant, physically separated and they can supply each of the power distribution systems of classes I, II and III.

Classes I and II are sized to supply power to selected safety-related loads before needing a connection to classes III, IV or other external power source.

IV.11. PLANT LAYOUT

CAREM Reactor Pressure Vessel (RPV) is placed inside a pressure suppression containment system, which contains the energy and prevents fission product release in the event of accidents ***(Figure IV-5)***.

The building surrounding the containment is placed in a single reinforced concrete foundation mat. It supports all the structures with the same seismic classification, allowing the integration of the RPV, the Safety & Reactor Auxiliary Systems, the Fuel Elements Pool and other related systems in one block.

The plant layout is divided in three main areas: Nuclear Module, Turbine Module and Control Module.

IV.11.1. Nuclear module

This building acts as a Secondary Containment. The Containment itself is a free standing, vertical, cylindrical reinforced concrete structure, with flat head and bottom, designed to support pressure and temperature conditions, and acts as a barrier to prevent fission product release to the Secondary Containment in the event of an accident.

The Nuclear Module has another relevant structural component in the shape of a box surrounded by 6 levels. In the upper part of this box are the Fuel Elements Pool and the Auxiliary Pool, and in the lower part are mainly Safety-Related Sistems equipment. In CAREM 25 these six levels are described below from 11.1.1 up to 11.1.6:

IV.11.1.1. Level +15.20: It is the Reactor Hall where tasks related with the refuelling will be performed.

IV.11.1.2. Level +10.00: The Exhaust stage of the HVAC system (Heating, Ventilation /Air Conditioning), the shielded rooms for filters and resins beds of the several Water Purification Systems and the Gaseous Waste Treatment System.

IV.11.1.3. Level +5.20: Valves rooms for the filters and resins beds, electrical swichboards and Standby Gas Treatment System.

IV.11.1.4. Level 0.00: The connection between the Control and Turbine Modules is here, as well as the emergency exit and the access to the Emergency Injection Systems. workshops, compressors, chilled water for HVAC, electrical swichboards and transformers.

IV.11.1.5. Level -5.80: All the liquid effluents and spent resins collected are stored in shielded pools and treated at this level. Also the process equipment for the Reactor Auxiliary Systems like pumps and heat exchangers are housed in this area, with physical separation of equipment belonging to different redundancies, HVAC Injection Equipment and Radiactive Liquid Effluents Equipment.

IV.11.1.6. Level -10.10: Pumps, filters and heat exchangers of Safety-Related Systems are housed in this area.

IV.11.1.7. Containment: The containment is divided into two main compartments: a Drywell and a Wetwell.

The Upper Drywell (+10.00 m) lodges the Second Shutdown System, the relief valves and the headers of the Residual Heat Removal System. The Emergency Condenser Pool is located at this level. The Central Drywell houses the RPV and below it, separated by shielding, is the Lower Drywell. The Peripheral Drywels surrounds the Central Drywell. The Peripheral Drywell at +5.20 m houses independent HVAC pieces of equipment and Safety-Related Valves. The Peripheral Drywell at 0.00 m houses the pipelines connected to the SG’s. The Wetwell (below the Peripheral Drywell, and surrounding the Central Drywell) is partially filled with water, conforming the Pressure Suppression Pool. The tube vents are immersed deep in the pool and they connect to the floor of the Peripheral Drywells. The Lower Drywell, at -10.10 m, houses Safety-Related systems equipment.

IV.11.2. Turbine module

It houses the Turbo-Generator Group, auxiliary services such as De-mineralised Water Production Ssystem, chilled water, service steam, condensate polishing, and electric switchboards.

Close to the Reactor Module are the redundant diesel-generators with switchboards.

IV.11.3. Control module

The Control Complex is placed in this area of the building. It is formed by the Cable Room, Instrumentation Rooms, Main Control Room, the Remote Shutdown System (or Secondary Control Room), the Heating, Ventilating and Air Conditioning System for each Control Room and the offices for the operation personnel.

Rest rooms, lockers, Radiological Protection Facilities and the Safety Access System for personnel access lock to the Nuclear Module, are located in the basement.

IV.12. NUCLEAR SAFETY

Emphasis has been given since the design genesis to prevention of core degradation accidents by means of passive safety features, guarantying no need of active systems or operator actions for a period of at least 36 hours.

Technical and safety advantages are obtained with the CAREM design compared to the traditional design:

In order to simplify the design, the whole high-energy primary system, core, steam generators, primary coolant and steam dome, are contained inside a single pressure vessel. This considerably reduces the number of pressure vessel penetrations and simplifies the layout. Due to the absence of large diameter piping associated with the primary system, no large LOCA has to be handled by the safety systems. This integrated concept and natural convection, have several advantages, such as:

- Elimination of large LOCA that considerably reduce the needs in ECCS components, AC supply systems, etc.
- Eliminating primary pumps precludes loss of flow accidents.
- The development of innovative hydraulic mechanism completely located inside the reactor pressure vessel eliminates the rod ejection accident.
- Large coolant inventory in the primary results in large thermal inertia and long response time in case of transients or accidents.
- The large water volume between the core and the wall leads to a very low fast neutron dose over the RPV wall.
- Shielding requirements are reduced by the elimination of gamma sources of dispersed primary piping and parts.
- The ergonomic design and layout make the maintenance easier. Maintenance activities such as the steam generator tubes inspection does not compete with refueling activities because it will be carried out from outside the vessel.
- The use of less active components increases plant availability and load factor, reducing the frequency and kind of initiating events.

IV.13. SAFETY SYSTEMS AND FEATURES

The safety systems are duplicated to fulfil the redundancy criteria ***(Figure IV-6)***. The shutdown system should be diversified to fulfil Argentine Regulatory Body requirements.

The *First Shutdown System (FSS)* is designed to shut down the core when an abnormality or a deviation from normal situations occurs, and to maintain the core sub-critical during all shutdown states. This function is achieved by dropping neutron-absorbing elements into the core by the action of gravity.

Hydraulic Control Rods Drives (CRD) avoid the use of mechanical shafts passing through RPV, or the extension of the primary pressure boundary, and thus eliminates any possibilities of large Loss of Coolant Accidents (LOCA) since the whole device is located inside the RPV. Their design is an important development in the CAREM concept. Nine out of twenty-five CRD (simplified operating diagrams are shown in ***Figure IV-7***) are the Fast Shutdown System.

The Second Shutdown System (SSS) is a gravity-driven injection device of borated water at high pressure. The system consists of *an assembly of* two tanks located in the upper part of the containment. *The assembly* is connected to the reactor vessel by two piping lines: one from the steam dome to the upper part of the tank, and the other from a position below the reactor water level to the lower part of the *other* tank. When the system is triggered, the valves open automatically and the borated water drains into the primary system by gravity. The discharge of *the assembly* produces the complete shutdown of the reactor.

The Residual Heat Removal System (RHRS) has been designed to reduce the pressure on the primary system and to remove the decay heat in case of Loss of Heat Sink. It is a simple and reliable system that operates condensing steam from the primary system in emergency condensers. The emergency condensers are heat exchangers consisting of an arrangement of parallel horizontal U tubes between two common headers. The top header is connected to the reactor vessel steam dome, while the lower header is connected to the reactor vessel at a position below the reactor water level. The condensers are located in a pool filled with cold water inside the containment building. The inlet valves in the steam line are always open, while the outlet valves are normally closed, therefore the tube bundles are filled with condensate. When the system is triggered, the outlet valves open automatically. The water drains from the tubes and steam from the primary system enters the tube bundles and condenses on the cold surface of the tubes. The condensate is returned to the reactor vessel forming a natural circulation circuit. In this way, heat is removed from the reactor coolant. During the condensation process the heat is transferred to the water of the pool by a boiling process. This evaporated water is then condensed in the suppression pool of the containment.

The Emergency Injection System prevents core exposure in case of LOCA. The system consists of two redundant accumulators with borate water connected to the RPV. Tanks are pressurised, thus when during a LOCA the pressure in the reactor vessel reaches a relative low pressure, rupture disks break and the flooding of the RPV starts, preventing core un-covery for a long period. The R*esidual Heat Removal System* is also triggered to help to depressurise the primary system, in case the breakage area is small.

Two safety relief valves protect the integrity of the reactor pressure vessel against overpressure, in case of strong unbalances between the core power and the power removed from the RPV. Each valve is capable of producing 100% of the necessary relief. The blow-down pipes from the safety valves are routed to the suppression pool. An operated relief valve system has been designed to depressurize the primary system in case of failure of RHRS.

The primary system, the reactor coolant pressure boundary, safety systems and high-pressure components of the reactor auxiliary systems are enclosed in the primary containment - a cylindrical concrete structure with an embedded steel liner. The primary containment belongs to the pressure-suppression type with two major compartments: a drywell and wetwell. The drywell includes the volume that surrounds the reactor pressure vessel and the second shutdown system rooms. A partition floor and cylindrical wall separate the drywell from the wetwell. The lower part of wetwell volume is filled with water that works as the condensation pool, and the upper part is a gas compression chamber.

A summary of functions to cover and the available safety systems is shown in ***Table IV.I.***

Table IV.I. Safety Functions and Safety Systems

Safety Function	Safety System
Reactivity Control	First Shutdown System: Safety control rods Second Shutdown System: Boron Injection
Primary Pressure Limitation	Safety Relief valves Residual Heat Removal System
Primary Depressurisation	Residual Heat Removal System Operated Relief Valve
Primary Water Injection	Low pressure: Emergency Injection System Second Shutdown System
Secondary Pressure Limitation	Relief valves
Residual Heat Removal	Residual Heat Removal System

For CAREM 25 accident analysis several initiating events were considered:

Reactivity insertion accident: as the innovative Hydraulic Control Rod Drive for the Fast Shutdown System and the Adjust and Control System is located inside the RPV, Rod Ejection Accident is avoided, only inadvertent control rod withdraw transients are postulated. Two scenarios considering FSS success and FSS failure with SSS actuation were modelled assuming conservative hypothesis. Simulation results show that safety margins are well above critical values (DNBR and Critical Power Ratio), no core damage is expected. Moreover, as there is no boron in the coolant, boron dilution as reactivity initiating event is precluded.

Loss of heat sink: in case of a total loss of feedwater to the steams generators, the Residual Heat Removal System is demanded cooling the primary system reducing reactor pressure to values lower than the ones of hot shutdown. In case of hypothetical failure of FSS, the reactor power reduces due to the negative reactivity coefficients without compromising the fuel elements. The SSS will guarantee medium and long-term reactor shutdown.

Total loss of flow: due to the absence of primary pumps, this initiating event is avoided.

Loss of coolant accident: RPV penetration maximum diameter is limited by design, therefore no large LOCA is possible and there is no need of a high-pressure injection system. In case of LOCA the FSS, SSS, RHRS are demanded and when pressure decreases the *Emergency Injection System* discharge water to keep the core covered for several days. As no credit is given by design to active systems, the secondary system is not considered to cool and depressurise the primary system in safety evaluations, of course if it is available and in case of need it could be used as part of Accident Management Strategy. Moreover, by design no credit is given to a broken pipe as an injection line (steam coming into the RPV from the containment in case of high depressurisation of the primary system due to the use of the steam generators). The reactor inherent response to LOCA was also analyzed, considering FSS success and failure of all the Safety Systems related with core cooling. Due to the large water inventory over the core and the small penetration diameters through the RPV, the core uncovers after several hours.

Steam generator tube rupture: this accident is mitigated by isolating the group of steam generators affected, closing both the steam and feedwater lines. The secondary side of the steam generators reaches thermal equilibrium with the primary circuit, equalising pressure with this system. Eventually the reactor could continue operating at 50% of power.

Steam line break accident: The sudden depressurisation of the secondary side of the steam generators increase heat removal from the primary system with the consequent core overpower. Reactor shutdown (FSS and SSS) and Residual Heat Removal System are demanded and the reactor reaches a safe condition. In case of hypothetical failure of both shutdown systems, reactor overpower does no compromise safety critical values (DNB and CPR) because primary total heat removal by the steam generators is intrinsically limited by the reduced tube side water inventory.

Blackout (SBO): It is one of the events with major contribution to core meltdown probability in a conventional light water reactor. The extinction and cooling of the core and the decay heat removal are guaranteed without electricity by the passivity of safety systems. Loss of electrical power produces the interruption of the feed-water to the hydraulically driven CRDs, and thus produces the insertion of the absorbing elements into the core. Nevertheless in case of failure of the First and Second Shutdown Systems (both passive), in CAREM, feedback coefficients will produce the self-shutdown of the nuclear reaction without compromising safety related variables. The decay heat is removed by the Residual Heat Removal System with autonomy of several days.

As a general conclusion, it could be said that, due to the large coolant inventory in the primary circuit, the system has large thermal inertia and long response time in case of transients or accidents.

Severe accident prevention and mitigation features: The CAREM concept highly enhances accident prevention and mitigation by simplicity, reliability, redundancy and passivity. Nevertheless, in case of the extremely low probability of failure of the passive safety systems (both redundancies) or no recovery actions after the design period to be covered by the passive safety systems (grace period of at least 36 h), a severe accident could be postulated to occur. Several features are considered to protect the confinement and to manage hypothetical severe accidents, allowing also the optimum use of all process systems for the primary cooling system and containment recovery after the grace period.

- The absence of large LOCA prevents an early and sudden containment pressurisation, and together with the impossibility of a high reactivity insertion (no rod ejection) the possibility of a fast core melt and early containment pressurization are limited.

- Complementary and simple measures and accident management after the design period to be covered by the passive safety systems.

- Prevention of high-pressure core melt situation is ensured by means of the Residual Heat Removal System, complemented by relief valves opening.

- The suppression pool cooling and purification system cools and refills –if necessary– the suppression pool and the cooling pool for the residual heat removal system and feeds spray in the dry and wet-well to depressurise the containment. In the event of a LOCA, this system is capable of feeding pure water into the RPV.

- Devices for reduction of the hydrogen-concentration in the containment.

- The suppression pool type containment provides a good physical mechanism for fission products retention by water.

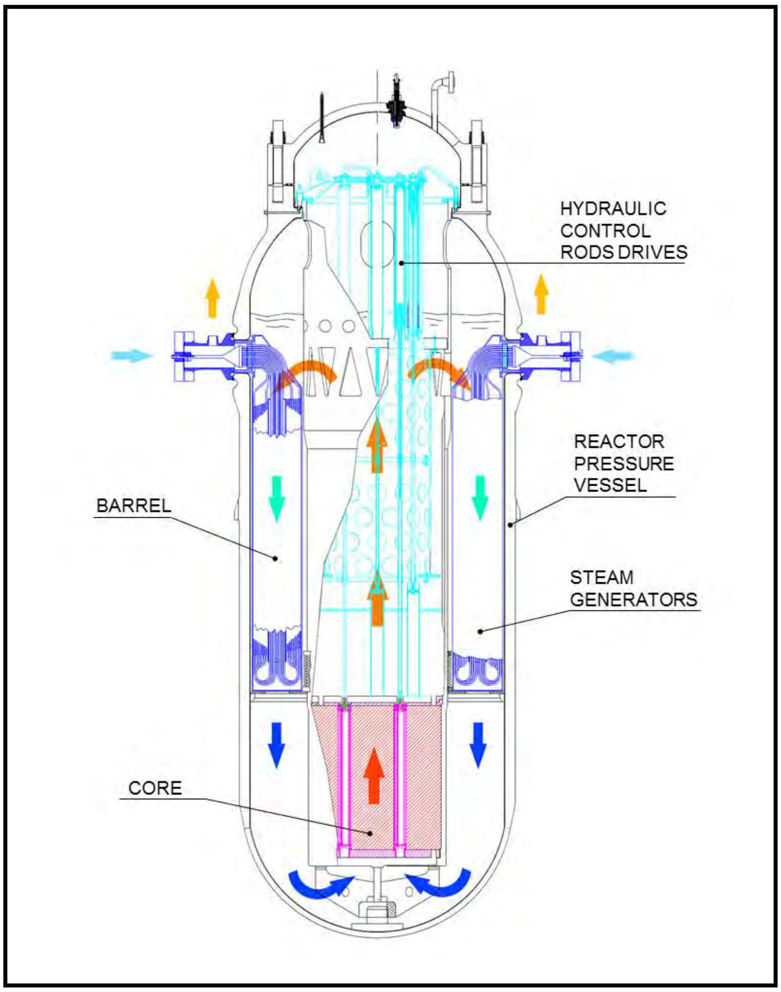


Figure IV.1 - Reactor Pressure Vessel

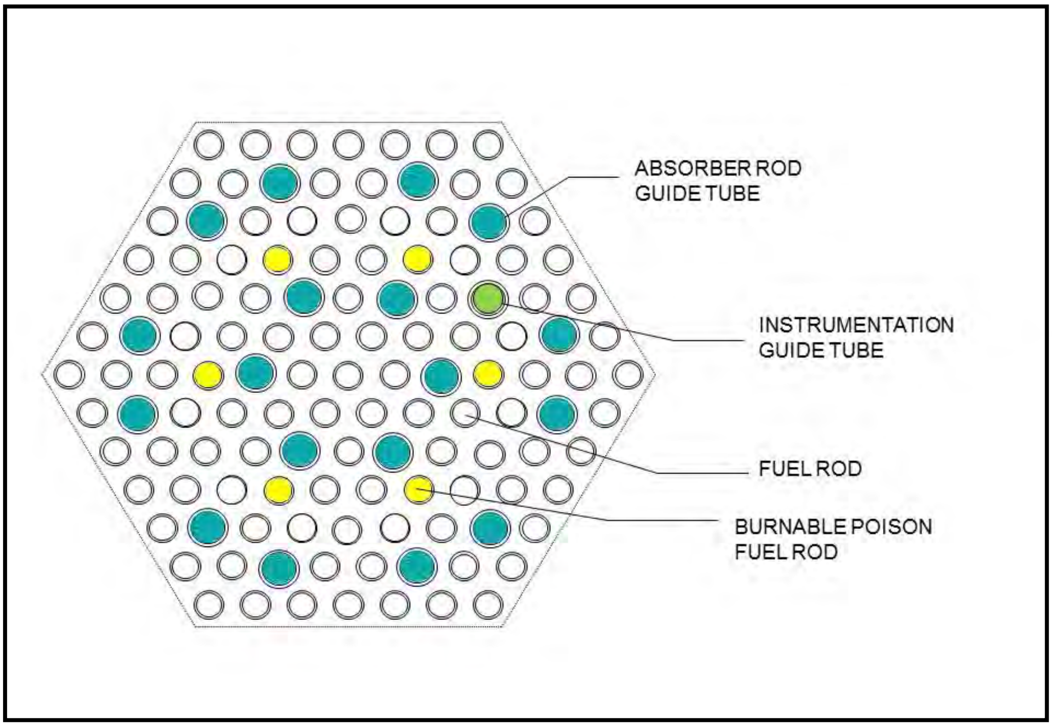


Figure IV.2 - Fuel Assembly Diagram. Fuel rods, guide thimbles and instrumentation thimble distribution

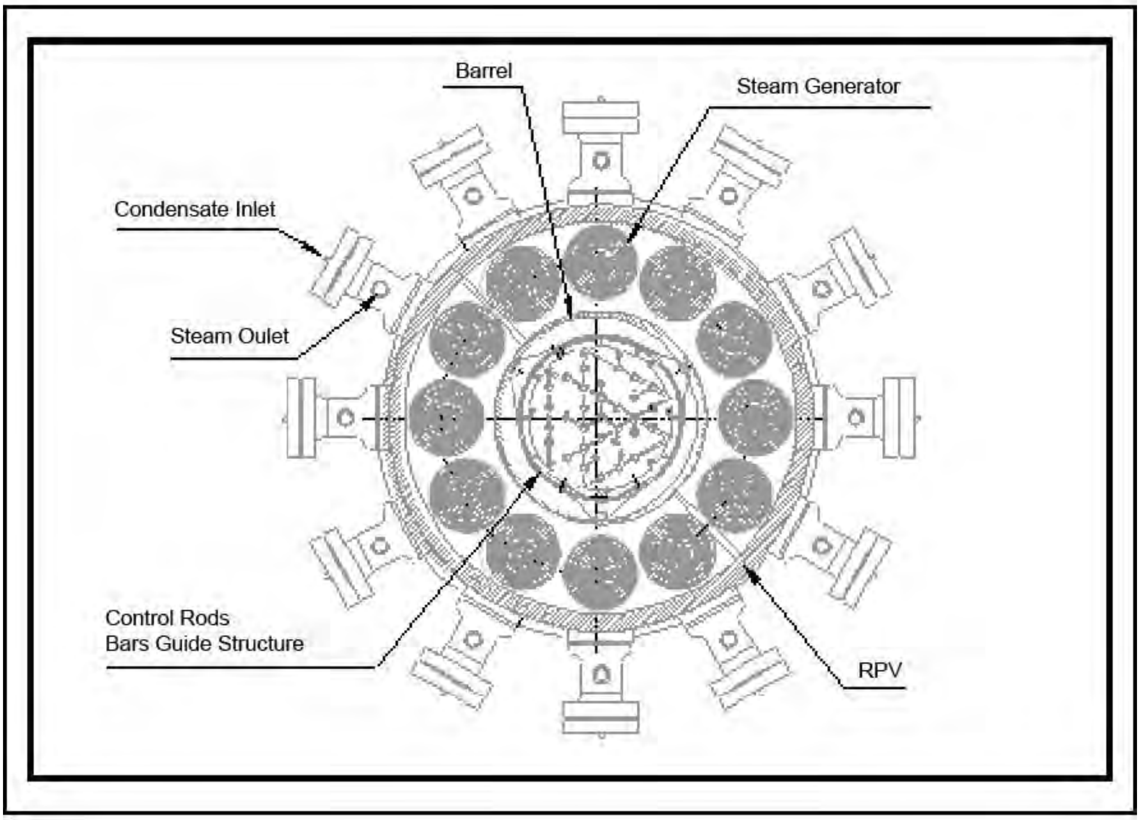


Figure IV.3 - Steam Generation Layout

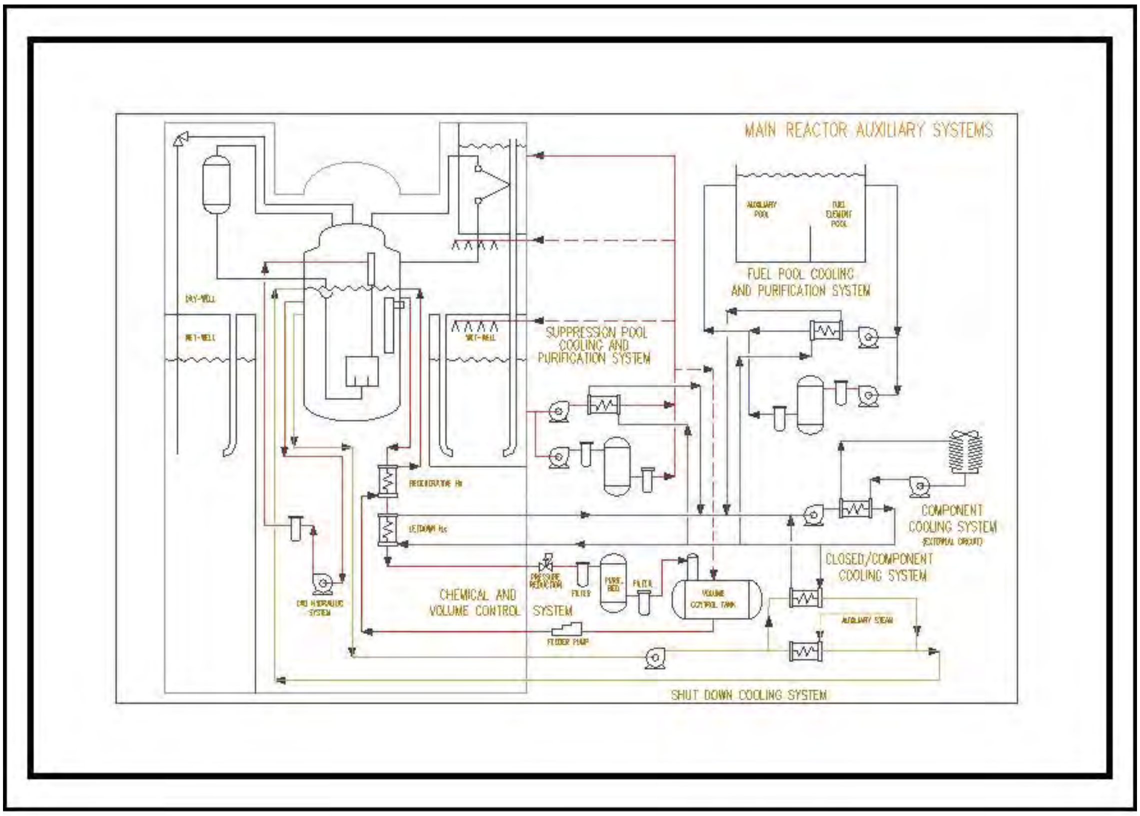


Figure IV.4 - Auxiliary Systems

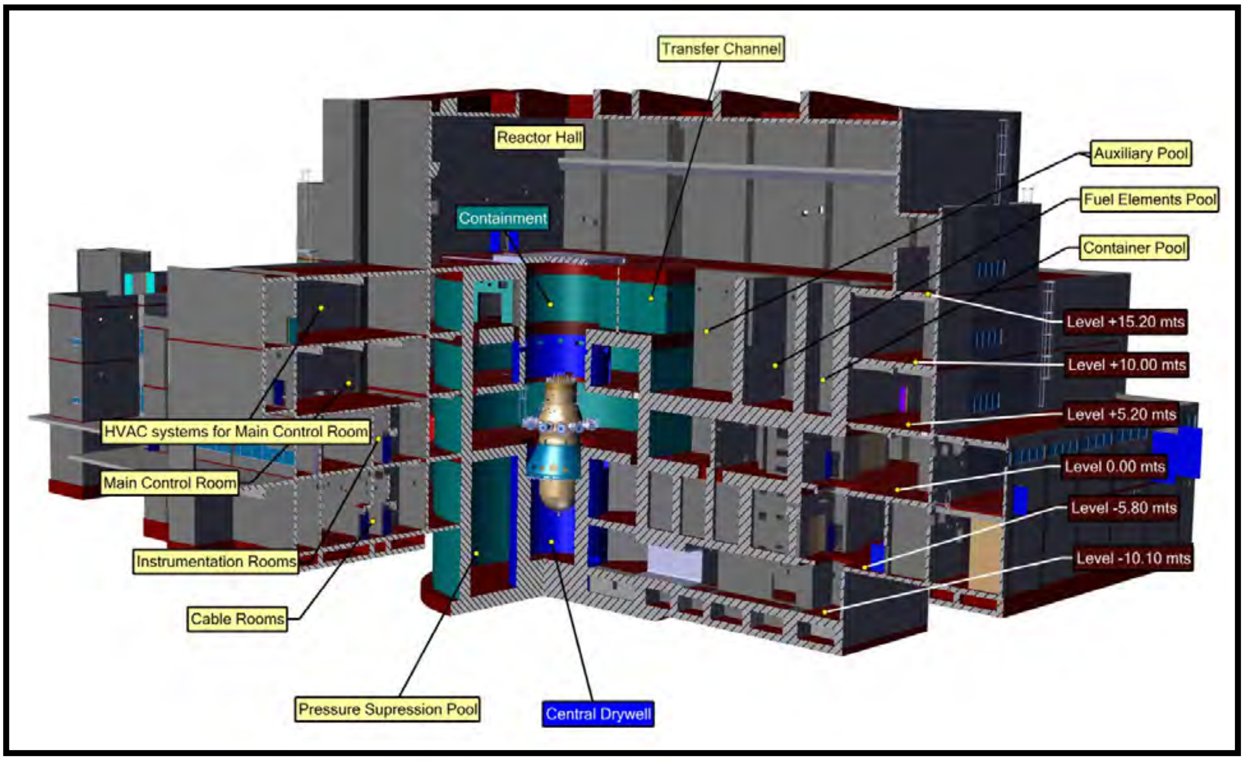


Figure IV.5 - Plant Layout

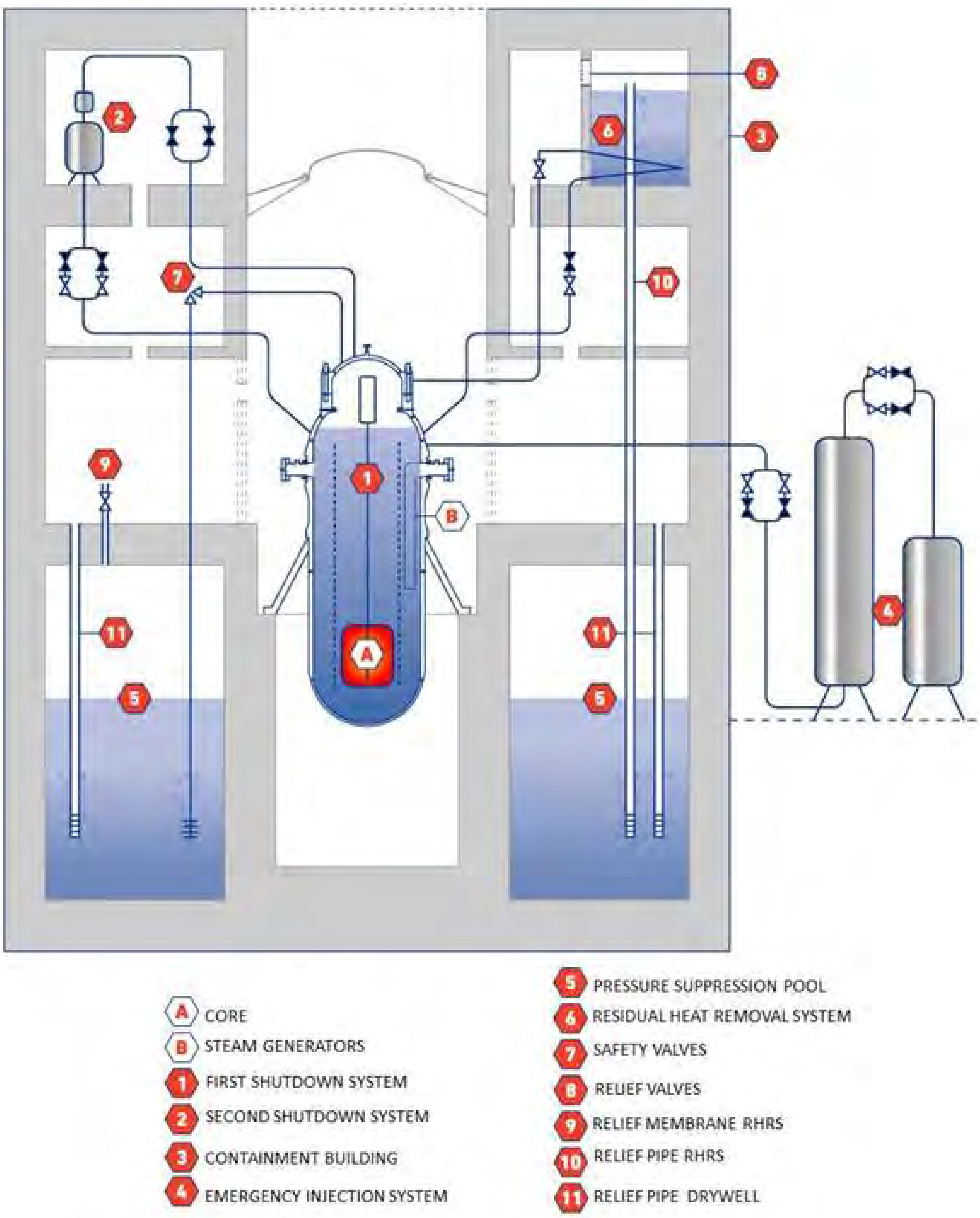


Figure IV.6 - Containment and Safety Systems

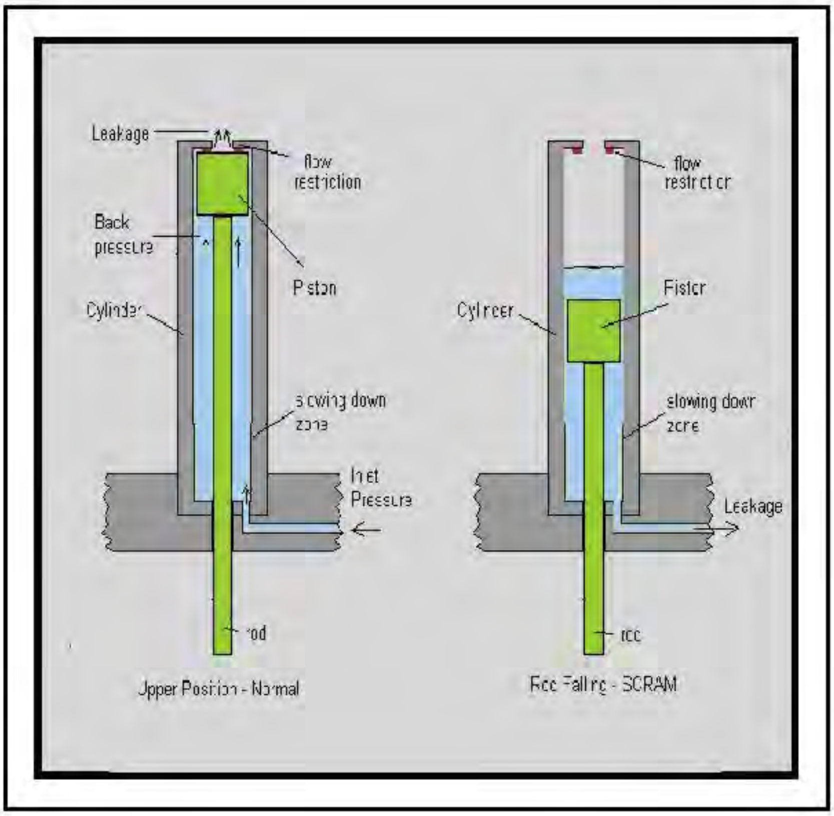


Figure IV.7 - Simplified Operation Diagram of a Hydraulic Control Rod Drive (Fast Shutdown System)

ANNEX V

EXAMPLES OF LESSON LEARNED AND CORRECTIVE ACTIONS RESULTING FROM NATIONAL AND INTERNATIONAL OPERATING EXPERIENCE AND EVENTS

The Operative Experience derived from National and International events are given in this Annex.

The most significant operational events in CNA I, CNA II and CNE during the period March *2016* – March *2019*, are listed and how the licensees and the ARN consequently and accordingly acted, are shown.

V.1. EXAMPLES OF LESSON LEARNED AND CORRECTIVE ACTIONS RESULTING FROM INTERNATIONAL EVENTS AND OPERATING EXPERIENCE

The examples of Lessons Learned/Corrective Actions listed below, were implemented by the plants in the report period.

EVENT - OPERATING EXPERIENCE	EXAMPLES OF LESSONS LEARNED - CORRECTIVE ACTIONS
Worker receiving additional radiation due to radioactive source shook out by mistake during radiographic test (Qinshan 2-4, PHWR, 2014-01-17)	*CNA I-II: Disseminate the event among radioprotection staff*
Opportunities realized through analysis of 2016 injury data trending (Bruce a Unit 1, PHWR, 2017-01-17)	*CNE: Issue of a report analysing accident and health and safety indicators for 2015-2017.*
Unplanned reactor shutdown and equipment malfunctions caused by water intrusion (Common event, -, 2009-04-14)	*CNA I-II: Incorporate the event failure hypothesis into the internal flood analysis.*
Emergency shutdown and spurious safety injection (Tihange 1, PWR, 2015-12-18)	*CNA I-II: Inclusion in Operational Annual Retraining.*
Emergency core cooling system signal actuation (Sequoyah 1, PWR, 2016-02-09)	*CNA I-II: Inclusion in Operational Annual Retraining.*
Loss of 500 kV line during 5113 2gb switch alignment (Embalse, PHWR, 2016-05-19)	*CNA I-II: Write a formal protocol about communications between NA-SA and TRANSENER (grid operator). This protocol must be applicable for CNE and CNA in order to improve actual communication*
Event involving the generation of static electricity in main transformer at02 due to operation of oil motor pumps (Trillo 1, PWR, 2015-05-19)	*CNA I-II: Disseminate the event in Electrical Maintenance.* *Verification of main transformers (AT, BT, BS and BAT) work orders having instructions for grounding.*
Incomplete filling of mineral fiber in fire doors (Neckar 1, PWR, 2013-05-23)	*CNA II: Write a fire door periodic test.*

Loss of environmental qualification of rosemount pressure transmitters caused by a broken neck seal (Doel 1-4, PWR, 2016-07-08)	*CNA I-II: Disseminate the event in I&C.* *Write a procedure related to how to perform maintenance tasks form components with environmental qualification.* *Include the event in the next 5 following work packages related to tasks in components from the environmental qualification master list.* *CNE: Issue of a work order to verify Rosemount transmitters in shutdown system #1 and #2 and in emergency core cooling system.*
Fire system pipework failure after cryogenic isolation (Sizewell b1, PWR, 2017-05-03)	*CNA I-II: Inclusion of event recommendations into the new freezing procedure.*
Bypasses in the volumetric protection could cause flooding from the turbine hall to the electrical building (Bugey 2, PWR, 2017-06-08)	*CNA I-II: Incorporate the event failure hypothesis into the internal flood analysis.*
External body contamination of a worker (Chinon b-2, PWR, 2016-04-10)	*CNA I-II: Disseminate the Event in the Radioprotection annual training.*
Electric arc flash accident resulting in minor hand burns (Cofrentes, BWR, 2017-10-06)	*CNA I-II: Disseminate the Event within Industrial Safety and Electrical Maintenance Staff.*
Unplanned reactor trip from 100% as a result of the loss of a 4 kV BUS (Turkey Point 3, PWR, 2017-03-18)	*CNA I-II: Include the event in the Industrial Safety annual training.*
A group industrial accident (Leningrad 3, LWGR, 2017-03-15)	*CNA I-II: Include the event in the Industrial Safety annual training.*
Two workers lost consciousness in a confined space due to lack of oxygen during assembly of a liquid waste system valve (Ascó 2, PWR, 2017-10-26)	*CNA I-II: Include the event in the Industrial Safety annual training.*
Increase of water hyacinth in river and down power (Atucha 2, PHWR, 2016-01-22)	*CNE: The Plant has initiated an evaluation for the rotating filter chains loss of tension detection system*
Failure to establish and maintain required reactivity shutdown margin following a reactor scram (Ningde 3, PWR, 2015-06-24)	*CNE: Modification of Operating Technical Specifications specifying the time in subcritical state before stablishing guaranteed shutdown. Inclusion in Operational Annual Retraining*
Reactor taken from power operations to cold shutdown conditions as a result of on-site flooding (Fessenheim 1, PWR, 2014-04-09)	*CNE: Issue of a work instruction for grating cleaning and drainage visual inspections.* *CNA I-II: Incorporate the event failure hypothesis into the internal flood analysis.*
Fuel channel specified on fuelling list could not be refueled due to high liquid zone level resulted in a reactivity management event (Pickering A4, PHWR, 2017-05-15)	*CNE: Modification of Operating Technical Specifications specifying the time in subcritical state before stablishing guaranteed shutdown. Inclusion in Operational Annual Retraining*
Automatic reactor trip due to high flux on source range channel while the reactor was being taken critical (Belleville 1, PWR, 2017-10-22)	*CNE: Inclusion in Operational Annual Retraining*

V.2. MOST SIGNIFICANT OPERATIONAL EVENTS IN CNA I, CNA II AND CNE. LESSONS LEARNED AND CORRECTIVE ACTIONS RESULTING FROM NATIONAL EVENTS AND OPERATING EXPERIENCE

V.2.1. Reactor trip due to moderator pump low flow signal. (CNA II, September 14th, 2016)

On 14/09/2016, during normal power operation in unit 3, three successful reclosing of 500 kV line were produced. After eighteen minutes from the last reclosing, another one unsuccessful occurred, in which voltage was not completely recovered in the secured BME busbars that feed the moderator pumps (380 V). Three seconds later, the line disconnection was produced and, in this case, the voltage in the busbars descended under 304 V causing the trip of the four moderator pumps (JFB10/20/30/40AP001) due to undervoltage and announcing moderator pump low flow signal, which cause SCRAM and turbine trip. After approximately a second, when the voltage was recovered in BM busbars, the four moderator pumps were automatically reconnected.

Direct Cause:. Under voltage, voltage breakdown.

Immediate Actions: The plant was on hot shutdown for 49 hours.

Root Cause: Original design inadequate. The reactor trip was due to an interlock in the reactor protection system. This interlock produces a SCRAM when the flow of the two circuit of the moderator is lower than 180Kg/s. This kind of situation occurs when low voltage in moderator pump feeding busbars is detected and generates a signal which disconnects the mentioned pumps momentarily. Due to the logical used, although voltage is quickly recovered, the reconnection time never last less than a second.

Lessons learned / Corrective actions:

- *Create a design modification to avoid the SCRAM during the one second stop of the moderator pumps.*
- *Request the designer to a possible optimization of the voltage regulator parameters, in order to keep voltage generator and busbars, as stable as possible when these transients in 500kV lines occur.*
- *Ask CAMMESA (Electrical Energy admin company) for a meeting with the objective to ask them the correspondent actions to keep 500kV line stability.*

V.2.2. Deformation of the Guide Tubes of the Neutron Flux Probe N2 and N3 (CNA II, April 21st, 2017)

During the moderator´s tank inspections of the Atucha II Reactor, buckling was observed in the guide tubes (GT) of the neutron flux probe N2 and N3. The causes of the jamming between the GT hub of the probes and the orifice of the moderator tank lid are related to the design of the GT.

Direct Cause: The jamming on top of the GT occurred during the transition from hot state plant condition to cold shutdown. When the reactor is cooled, there is a clogging between the upper bushing of the GTs and their respective housings in the tank lid of the moderator. When contracting and not being able to slip into the hole, the GTs buckled.

Immediate Actions: The extraction of the N2 probe was carried out. The functionality of the related neutron flux detectors was replaced making adjustments in the regulations of the control and distribution of power, and in the Limitation System. Engineering studies were also carried out to ensure that the N3 will keep its structural integrity.

Root Cause: Design.

Lessons learned / Corrective actions:

- *Incorporate the inspection of the N3 probe status in the annual Reactor Internal Inspection Program*
- *Start the actions to change the design of the GT of the N probes.*

V.2.3. Damaged Hydraulic Rods Guide Tubes (CNA II, April 21st, 2017)

During the inspections of the moderator tank, that were performed during the present scheduled outage, there was observed a break in the hydraulic rod guide tubes SX1 and SX3 and holes at the level of the skids of the hydraulic rod guide tube SH2. The original design had contemplated the use of 9 hydraulic control rods to regulate the xenon oscillations and as shutdown negative reactivity during the outages. Studies before the assembly of the reactor internals established that the hydraulic control rods were not necessary for the operation of the reactor. The hydraulic rod guide tube was assembled but the internal rod was not assembled due to this change of design criteria.

Direct Cause: It was determined that, given the way the assembly was performed, the heavy water entering from the lower plenum, passes into the moderator tank through the holes in the outer tube, causing that flow vibrations that rub the centering against the outer tube of the guide tube causing wear until it is drilled. This damage is progressive and decreases in the flow direction of the heavy water flow.

Immediate Actions: Prior to the restart of the plant, all hydraulic rod guide tubes were sealed according to a permanent Installation Modification.

Root Cause: The original design document indicated that, in case the control rods were not mounted, a "dummy" rod or a plug should be mounted in its place to reduce the flow of heavy water flowing through the guide tube. This "dummy" rod was not mounted. In addition, the pipes that exit at the top are sealed and no water comes out of them.

V.2.4. Air leak by a High Pressure Emergency Coolant Injection System air accumulator tank safety valve (CNA I, June 13th, 2018)

On 13/06/2018, with unit I in full power operation, a pressure drop in the high pressure emergency coolant injection system air accumulator tank, belonging to one of the two loops, occurred twice due to air leakage in the tank safety valve. The leak occurred due to a progressive deterioration of the valve seat induced by pressure waves of 52 kg/cm^2 produced during the system surveillance tests. The pressure was higher than the valves set value and produced vibrations which progressively damaged the valve seat. The test procedures did not consider this issue. The affected loop was unavailable for 18 hours due to corrective maintenance.

Direct Cause: Due to system compressors were not able to recover normal pressure value, and to achieve that condition reserve air compressor had to be started-up manually, the immediate reparation of the TJ50S92 valve was performed. To do this, it is necessary isolate and depressurized TJ50B01 tank.

Immediate Actions: A temporary instruction of NZ62 surveillance tests was issued. It indicates to perform the tests with pressure low values and with the SEMPELL surveillance device pneumatically isolated in order not to withdraw the TJ50S92 and TJ60S92 valves magnetic overload in case of pressure transient in the pipes. Devices were installed into the system in order to register the behaviour during NZ62 surveillance tests.

Root Cause: After the NZ62 surveillance tests, the pressure wave and pipe vibrations affected progressively the TJ50S92 valve seat leading the leakage

Lessons learned / Corrective actions:

- *Evaluate TJ50S92 and TJ60S92 valves trip adjustment tolerance in order to set it in the maximum admissible, reducing the negative tolerance.*
- *Perform the final modification to the protocol of the N62 surveillance test.*

V.2.5. Manual SCRAM after a moderator circuit valve failed to open (CNA I, October 19th, 2018)

On 19/10/2018 with unit I at full power a surveillance test of 'switching from moderator system to emergency cooling mode' had been performed but afterwards the operator could not return the moderator circuit valve to the normally open position and a manual SCRAM was undertaken in accordance with the tech specs. Caused by cables which were loose due to weld fatigue. These *cables produce mechanical stress on the welding; when the withdrawable unit is positioned in TEST mode, its front is displaced in order to allow the cabinet opening. This practice was recently implemented, after the Arc Flash study. There is a design flaw, since when displacing the panel, mechanical stress is transferred from the cables to the components.*

Direct Cause: It was determined that the cables were loose due to welding fatigue.

Immediate Actions: The plant was stabilized in hot shutdown condition.

Root Cause: These cables produce mechanical stress on the welding; when the WU is positioned in TEST mode, its front is displaced in order to allow the cabinet opening. This practice was recently implemented, after the Arc Flash study. There is a design flaw, since when displacing the panel, mechanical stress is transferred from the cables to the components. Besides, the cable is rigid and obsolete since it is a solid wire type and its diameter is 4mm2.

Lessons learned / Corrective actions:

- *Replace the solid wire cable with another more flexible and with a correct section in the Fiat valves withdrawable units (QM01S002/003/004/005/006/007 and QM02S002/003/004/005/006/007)*

V.2.6. Significant heavy water spill due to loss of ice plug with 100 hours outage extension and personal contamination (CNA I, December 30th, 2018)

On 31/12/18 unit 1 in cold shutdown heavy primary water (D_2O) spilled for 6 hours in the reactor building (9.5 tons lost) and the plant was moved to mid-loop status due to loss of ice plugs. To clean the clogged filters TA53/54N001 (D_2O supply of the main cooling pumps seal), two ice plugs were put in series using nitrogen. The filters stayed open because they have to be cleaned and not replaced, and the workers left the room due to high tritium activity. Ice plugs were without visual controlled and their losses were not detected for 2 hours. A worker was contaminated (3.6 mSv).

Direct Cause: Nitrogen dosage to ice plug was insufficient, causing heavy water loss.

Immediate Actions: The Shift immediately decreased the primary system pressure as a palliative action. The plant was taken to mid-loop status.

Root Cause: No ice plug and no ALARA risk analysis was done, no pre-job briefing was held and ice plug procedure adherence was violated. Lack of safety and radioprotection risk analysis and global lack of questioning attitude has led to significant loss of D_2O, 100 hours outage extension and worker contamination.

Lessons learned / Corrective actions:

- *Modify the “freezing procedure”, analysing actions to be taken for the different ‘risk types’ in order to minimize the risk and establish contingency actions.*
- *Re-qualify staff that makes the ice plugs, according to the modified freezing procedure, emphasising on observing the ice plug directly and by an alternative method.*
- *Disseminate the event among staff that makes the ice plugs, emphasising on the correct use of the freezing procedure and filling out of form (FO*129) of the freezing procedure.*
- *Include in the Mechanical Maintenance TA53/54N001 Instruction the recommendation to close the filters as soon as possible. In addition, add the use of component bagging and the use of a container with H_2O in order to dilute the drops that could spill when removing the filter and lid.*
- *Disseminate the event to shift crew supervisors of both units, emphasising that when necessary, they should establish together with the staff that makes the ice plugs a "Contingency Plan".*

V.2.7. Loss of 500 kV power supply during switchyard manoeuvres. (CNE, May 19th, 2016)

Summary: During a Refurbishment Outage with only external power supply for plant operation and while performing a switch alignment in the 500 kV switchyard, a wrong manoeuvre performed by the Transmission Grid Company (TGC) resulted in loss of offsite power. It resulted in loss of spent fuel bay (SFB) heat sink for 9 minutes.

Causes: The cause was that TGC shift manager jumped a few steps from the procedure. The other cause was inadequate communications between the station and TGC.

Lessons learned / Corrective actions:

- *To program training simulation for the Transmission Grid Company operators and Shift manager that contemplate similar scenarios.*
- *To reinforce the importance of external power supply in order to ensure nuclear safety.*
- *To implement an effective and permanent communication, via telephone, during every switchyard manoeuvre performed between Embalse and the Transmission Grid Company.*

V.2.8. Unexpected high concentration of tritium in the reactor building (CNE, November 17th, 2017)

During a refurbishment outage with the coolant and moderator heavy water drained and while the dry air injection was aligned to the moderator purification system to reduce dew point oscillations, a high concentration of tritium was vented from the reactor core internals. This resulted in contamination of the reactor building air, 522 workers received tritium dose higher than 0.27 mSv with five of them exceeding 5 mSv, and one worker received 10 mSv. The event is significant because it resulted in tritium dose to many workers with one worker receiving a dose of 10 mSv.

Causes: The cause was venting of tritium through an open pipe flange having only a plastic cover, which was broken due to overpressure generated by a wrongly closed valve of the Calandria air extraction system. This pipe was required to have a blank flange by procedure. The other cause was inadequate analysis before changing the air injection configuration and it was assumed that the pipelines that were being injected were already dry. No pre-job brief was performed and the radiological protection department was also not informed for tritium monitoring.

Lessons learned / Corrective actions:

- *Perform an Operation Decision Making (ODM) to evaluate the future actions in front of the configuration changes of the conservation system of the calandria.*
- *Re-install the continuous tritium monitoring system in the Reactor Building.*
- *Increase the tritium measuring routine frequency performed by the radiological protection officer in the moderator and calandria room.*
- *Install correspondent protections and closures on the pipes corresponding to the new systems connected to the calandria.*
- *Increase the monitoring frequency of the calandria's air extraction system components.*
- *Spread the event reinforcing the use of the human performance tools "Pre-Job Meeting" and "Use and Adherence to Procedures".*

V.2.9. Heavy Water Spillage in Reactor Room R-012 through ducts of Heavy Water Collection system (CNE, September 30th, 2018)

On September 28th, 2018, with Embalse NPP in refurbishment outage with ongoing filling task of heavy water into the Heat Transport and Moderator system, wanted heavy water flow could not be achieved during manoeuvres of new ion-exchange resin deuteration so it was decided to vent the pipeline in a different way to what is stablished in the Order To Operate issued for this task, this way required to align said pipeline to the Reactor's Heavy Water Collection system. This condition was not normalized after the task was performed and on September 30th a "Heavy Water Leak" alarm was triggered in the control room. Said leak spilled approximately 300 litres of heavy water into Reactor Room R-012 through ducts of Heavy Water Collection system which was later recollected.

Causes:

Direct Cause: Lapse on the alignment registration.

Root Cause: Lack of adherence to procedures.

Contributing Factors: weaknesses on the ion-exchange resin transport and deuteration and de-deuteration systems.

Lessons learned / Corrective actions:

- *Spread the event among the Operation shift crews reinforcing: Use and adherence to procedures; to keep a correct record of modifications on Orders to Operate; and the shift handover expectations.*
- *Through a Deficiency Report, create a device that allows connecting and draining the line into a drum. It is suggested that the validation of the design should be carried out by Operations and Maintenance to consider aspects of operational manoeuvrability and maintenance.*

V.2.10. Shutdown System #1 trip when a test procedure was being carried out (CNE, February 20th, 2019)

On 02/20/2019, Embalse NPP was carrying out Dynamic Commissioning Tests with the Reactor at 35% Full Power (FP).

During the execution of the test procedure, the Shut-Down System 1 (SDS#1) tripped by 2 out of 3 logic on Heat Transport System (HTS) High Pressure. The SDS#1 was tripped due to the simultaneous tripping of Channel D (trip generated by the test) and Channel F (spurious trip). Channel F spurious tripping was due to its own channel pressurization by a simultaneous fault in its valves, the test valve was not closed correctly and the process valve failed due to incorrect maintenance.

This situation caused a loss of power generation for 1 hour 22 minutes.

Causes:

Failure in test valve 68233-PV1F2 was due to damage to the valve seat. At the time of the event it could not be determined what caused the damage to the valve seat.

Failure in process valve 68233-PV1F1 was due to incorrect calibration due to lack of questioning attitude by maintenance when deciding on how to perform the calibration.

Lack of clear expectations of maximum deviation allowed in the HTS pressure indication.

The generated WO had an insufficient evaluation prior to its assignment to maintenance.

Weaknesses in keeping the MCR records updated.

Weaknesses in the Quality Control.

Lessons learned / Corrective actions:

- *Analyse and determine the pressure variations allowed for the HTS. Update the TP, Operating Manual and Monitoring Procedure with these values.*
- *Disseminate the event among the Operations shift crews, Instrumentation and Control Maintenance, and Quality Control reinforcing to keep the records updated and use the Human Performance Tools (questioning attitude).*

ANNEX VI

RESUME OF NA-SA CORPORATE QA MANUAL CONTENT

VI.1. TABLE OF CONTENTS

Introduction (overview, process approach).

Objective and scope.

Abbreviations and definitions.

Quality Assurance System:

- General requirements.
- Documentation requirements.

Management responsibility:

- Management commitment.
- Customer focus.
- Quality policy.
- Planning.
- Responsibility, authority and communication.
- Management review.

Resource Management:

- Provision of resources.
- Human resources.
- Infrastructure.
- Work environment.

Product realization:

- Planning of product realization.
- Customer-related processes.
- Design and development.
- Purchasing.
- Production and service provision.
- Control of monitoring and measuring equipment.

Measurement, analysis and improvement:

- General.
- Monitoring and measurement.
- Control of nonconforming product.
- Analysis of data.
- Improvement.

References.

Annex 1 Procedure matrix.

Annex 2 Nucleoeléctrica Argentina S.A. organizational chart.

Annex 3 Definitions.

Annex 4 Manual distribution list.

Annex 5 Quality policy.

Annex 6 SGC documents and records.

VI.2. INTEGRATED QUALITY AND ENVIRONMENT POLICY

Nucleoeléctrica Argentina S.A., a nuclear power plants operating company, committed to electric power generation in a safe, clean, efficient and competitive manner, establishing safety culture and transparency as core values of the organization, manifests and assumes the following Quality and Environment Policy:

- ***Ensure the control of activities***

  Perform a continuous effort to plan and control every activity of the Company directly or indirectly related to safety and the availability of its facilities, applying the appropriate codes and technical standards.

- ***Protect the environment***

  Make a continuous effort to prevent pollution and minimize the adverse environmental impact derived from our activities. Operate the facilities making a rational use of energy and natural resources.

- ***Continuously adapt the management to applicable regulations.***

  Comply with applicable regulations to different facilities and activities of Nucleoeléctrica Argentina S.A. and other requirements subscribed by the organization.

- ***Promote a management culture for risks and opportunities.***

  Evaluate the potential risks and opportunities of electric power generation activities and new projects.

- ***Promote staff training and knowledge management.***

  Train the staff ensuring that they are competent to perform the assigned tasks, to manage the risks of the activities under their control and to develop awareness of the impact of their tasks on safety and environment.

- ***Promote internal and external communications***

  Communicate our Quality and Environment Policy to the staff, ensuring their understanding and compliance. Make it available to interested parties. Inform about the benefits of the nuclear option and its contribution to the preservation of the environment.

- ***Continuously improve quality management and our environmental performance.***

  Strive for continuous improvement and effective management through the systematic and periodic evaluation of quality and environmental management, the implementation of detected improvement opportunities and the excellence practices of the international nuclear industry.

- ***Achieve customer satisfaction***

  Meet customers' requirements.

CONVENTION ON NUCLEAR SAFETY

Autoridad Regulatoria Nuclear
PRESIDENCIA DE LA NACIÓN