CONVENTION ON NUCLEAR SAFETY

NATIONAL NUCLEAR SAFETY REPORT

ARGENTINA - 2022

Autoridad Regulatoria Nuclear
PRESIDENCIA DE LA NACIÓN

NINTH REPORT

ARGENTINEAN NATIONAL REPORT FOR THE CONVENTION ON NUCLEAR SAFETY

August **2022**

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
Ninth Report

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CONTENTS

GLOSSARY

ABACC	*Agencia Brasileño-Argentina de Contabilidad y Control de Materiales Nucleares /* Brazilian-Argentine 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
CAs	Calculation Aids
CAAR	*Consejo Asesor de Aplicaciones de Radioisótopos y Radiaciones Ionizantes* / Advisory Council for the Application of Radioisotopes and Ionizing Radiation
CALPIR	*Consejo Asesor para el Licenciamiento de Personal de Instalaciones Relevantes /* Advisory Committe for the Licensing of Major Installation Personnel
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
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
CIEMAT	*Centro de Investigaciones Energéticas, Medioambientales y Tecnológicas* Center for Energy, Environmental and Technological Research
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

CONEAU	*Comisión Nacional de Evaluación y Acreditación Universitaria /* National Commission for University Accreditation
CONICET	*Consejo Nacional de Investigaciones Científicas y Técnicas /* National Council of Scientific and Technological Research
CONUAR	*Combustibles Nucleares Argentinos S.A.*
CPO	Crew Performance Observations
CPR	Corporate Peer Review
CRMRD	*Comité de Revisión de Modificaciones Relevantes al Diseño* / Relevant Design Changes Review Committee
CRT	*Comité de Revisión Técnica* /Technical Revision Commitee
CSA	Canadian Standards Association
CSS	Commission on Safety Standards
CVML	Calandria Vaut Make-up Line
CyMAT	*Comisión de Condiciones y Medio Ambiente de Trabajo* / Work Conditions and Environment for Public Sector Commission
DA	Design Authority
DBA	Design Basis Accidents
DBE	Design Basis Earthquake
DCC	Digital Control Computers
DEC	Design Extension Conditions
DFC	Diagnosis Flow Chart
DG	Diesel Generator
DiD	Defence in Depth
DNBR	Departure from Nucleate Boiling Ratio
DNPC	*Dirección Nacional de Protección Civil* / National Civil Protection Direction
DOE	U.S. Department of Energy
DSA	Deterministic Safety Analysis
ECCS	Emergency Core Cooling System
EduTA	Education and Training Appraisal
EECC	*Centro Externo de Control de Emergencias /* External Emergency Control Center
EFCVS	Emergency Filtered Containment Venting System
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
EPReSC	Emergency Preparedness and Response Standards Committe
EPRI	U.S. Electric Power Research Institute
EPS	Emergency Power Supply
EPZ	Extended Complementary Planning Zone
EQ	Environmental Qualifications
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 /* School of Engineering of Buenos Aires University
FOs	Field Operations
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
FSFs	Fundamental Safety Functions
FU	Follow-Up (mission)
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/Enhancement 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"*
IFMAP	Irradiated Fuel Management Advisory Programme
IGALL	International Generic Ageing Lessons Learned
IMPSA	*Industrias Metalúrgicas Pescarmona S.A.*
INA	*Instituto Nacional del Agua* / National Water Institute
INEEL	Idaho National Engineering and Environmental Laboratory
INES	International Nuclear Event Scale System
INPRES	*Instituto Nacional de Prevención Sísmica /* Seismic Prevention National Institute
INPRO	International Project on Innovative Nuclear Reactors and Fuel Cycles
INSAG	International Nuclear Safety Advisory Group
INVAP	INVAP S.E.
IPERS	International Peer Review Service
IRAM	*Instituto Argentino de Normalización y Certificación (ex Instituto de Racionalización Argentino de Materiales) /* Argentine Institute for Standarization and Certification
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	*Gerencia Licenciamiento y Control de Reactores Nucleares* / Licensing and Control of Nuclear Reactors Department
LISS	Liquid Injection Shutdown System
LLSFs	Low Level Safety Functions
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* / Argentine Accreditation Agency
OBE	Operating Basis Earthquake
OEP	Operating Experience Program
OIs	Operating Instructions
OJT	On-the-Job Training
OPDS	*Organismo Provincial para el Desarrollo Sostenible*
OPEX	Operating Experience
ORE	*Organización de Respuesta ante Emergencias* / Emergency Response Organization
OREE	External Emergency Response Organizations
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
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
SCST	Severe Challenge Status Tree
SDS	Shut–Down Systems
SDV	Screening Distance Value
SEDA	*Sistema de Evaluación de Dosis en Emergencia* / Accidental Dose Assessment System
SF	Spent Fuel
SFGs	Safety Functional Groups
SFP	Spent Fuel Pool
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
SIP	Safety Injection Pumps
SL	Seismic Level
SMA	Seismic Margin Assessment
SMN	*Servicio Meteorológico Nacional* / National Meteorological Service
SMR	Small Modular Reactor
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
TSG	Technical Support Group
TSM	Technical Support Mission
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 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* / San Juan National University
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	Vienna 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

CHAPTER 1
INTRODUCTION

This *Ninth* 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 the join CNS 8th and 9th Review Meetings with regard the progress made on challenges and suggestions from 7th Review Meeting, including those considered closed since the 8th Review Cycle have been taken into consideration. The National Nuclear Safety Report also contains information related to the Argentinean experience with response to COVID-19 Pandemic to ensure continued safety of NPPs.

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 *Ninth* National Nuclear Safety Report describes the actions that the Argentine Republic has carried out since the Eighth Nuclear Safety Report was issued (by middle 2019) until *2022*, 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 *6.8%* of the total electric power generated *in 2022 (up to May).*

CNA I (Atucha I Nuclear Power Plant) 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 with slightly enriched uranium (0.85%). The reactor is moderated and cooled with heavy water. *Since 2018, the plant is under long term operation.*

CNA II (Atucha II Nuclear Power Plant) 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 (Embalse Nuclear Power Plant) 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 refurbished and updated its licensing basis as part of the Life Extension project which was concluded in August 2019. Currently, is in normal operation.*

The CAREM prototype reactor (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. 24,804 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 on-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 the end of April 2019. *Operating License was issued in August, 2019.*

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 *Ninth* 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 *addressing the recommendations on the content of the Ninth National Report as stated by the Presidency of the Joint 8th and 9th Review Meetings.* 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 updated information on issues raised or requested by other countries at *the Seventh Review Meeting and proposals for suggestions and challenges identified in the draft Country Review Report for the 8th 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, 6th, 7th *and 8th* 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. *Besides, actions in response to COVID-19 Pandemic taken by the Licensee of operating NPPs are described.*
- 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. *Besides, actions in response to COVID-19 Pandemic taken by ARN are described.*
- 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.
- 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 operating 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:** *Draft Country Review Report for Argentina based on the Eight National Nuclear Safety Report.*
- **Annex II:** Answer to Questions or Comments - National Nuclear Safety Report - *2019*.
- **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 Corporate Quality Assurance 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 on-site 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.

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.

1.4.2. 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 November 16-17, 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 Prototype 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) for the 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 was 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.

Currently, ARN and NA-SA are working in the so called “PSR Basis Document” where the agreement on the general scope and requirements for the CNA II’s PSR, and its expected outcome, are documented. The PSR for CNA II will be submitted early in 2024.

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 detail in the Sixth and Seventh National Nuclear Safety Reports.

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, INCLUDING SAFETY ISSUES IDENTIFIED THROUGHOUT THE EIGHT REVIEW CYCLE

At the 7th Review Meeting in 2017, the Rapporteur resumed the challenges and planned measures to improve safety for Argentina on specific topics to follow-up on the Eight National Report. These topics are discussed in the following sections.

In addition, as encouraged by the Presidency of the Joint 8th and 9th Review Meeting, this chapter describes activities addressing new safety issues which were identified throughout the 8th review cycle.

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.

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 April 25th to 27th, 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 ARN´s 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 November 6-7, 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.*

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.

During the present reported period (2019-2022) the following activities were agreed within the ARN:

- *The IRRS mission originally set to October 2018 were postponed due to programmatic reasons to May 2020 and then, postponed due to the COVID pandemic;*
- *The IAEA´s SARIS methodology was applied for the full self-assessment previous to the IRRS mission, and all phases have been undertaken;*
- *The IRRS mission is scheduled from August 22nd to September 2nd, 2022.*

2.2. CHALLENGE 2: SALTO MISSION TO ATUCHA I

Argentina has selected for the methodological ordering of the Long Term Operation (LTO) project for CNA 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, CNA I received two PRE-SALTO missions, the last one took place during October 23rd to 31st, 2018.

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 CNA 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.

In addition to the above mentioned, the plant hosted a Pre-SALTO Follow-up mission during November 23rd to 26th, 2021, in order to IAEA review the progress in addressing each issue from 2018 Pre-SALTO mission. The review team concluded that the plant had progressed in resolving most of the issues: out of fifteen issues (15) from Pre-SALTO mission, two (2) are resolved and twelve (12) issues were assessed as "satisfactory progress to date". The remaining issue which is related to the long-term human resources plan that addresses the organizational requirements for LTO was assessed as insufficient progress to date.

Despite that there is still room for improving the activities for LTO, both the plant and the regulatory body, are strongly committed to strengthen preparation for safe LTO. The plant decided and so, communicated to IAEA, the decision to invite a full scope SALTO mission in 2023.

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 was required by the ARN as a means for retaining the corium in scenarios with extensive core damage. The strategy and its effectiveness were under analysis during the years 2016-2019. 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 firstly for CNA II NPP.*

Primarily, very detailed calculations with ANSYS / CFD were performed for SBO scenario, with conservative values for decay heat and history of relocation into the lower plenum. Since these analyses showed that the strategy was most likely unsuccessful, they were complemented by Best Estimate calculations to investigate if boundary conditions might have any effect. The calculations included 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.

The results of these analyses showed that the successfulness of the External Reactor Vessel Cooling strategy cannot be ensured in the whole set of severe accidents postulated in Atucha reactors.

The reason for this is that, even though the decay heat associated with relocated material in the lower plenum is low, Atucha reactors have a higher heat flux in RPV walls than those usually calculated for PWR reactors. Atucha's RPV lower plenum is filled with a massive steel structure in the shape of a hemisphere (usually called "filling body"). These structures give relocated corium rather low-in-height cylinder-like structure. Heat transfer and natural circulation patterns is therefore different in Atucha reactors than in PWRs and this is the main reason why analysis had to be performed in detail.

Analysis showed that in the long term, passive heat removal may not be sufficient to decrease RPV wall temperature, and after a few hours, integrity of the RPV could not be assured, since creep failure at high temperature was a threat

In practice, this countermeasure has been ruled out of the SAM Program.

CNA I - Corium-Barrier to improve Containment long term integrity

During the last years, NA-SA started to assess possible stabilization of molten material inside sump, to avoid containment breach due to Containment bypass through SIP suction lines, or at least delay to it, so as to decrease consequences in public as far as reasonable achievable. This task is being performed jointly by Safety Analysis, Engineering and Life Extension Project groups.

In CNA I, the sump design, volume, relative location of RPV and SIP suction lines, is such that in the event of a Severe Core Damage accident, if total meltdown of the core plus main internals is considered, the volume of corium could be contained inside a set of newly installed walls that would avoid direct contact with SIP suction lines.

The material specification of such walls, as well as the determination of expected integrity is under development by NA-SA and CONICET (specialists in ceramics materials have been contacted). The preliminary proposal is to construct this Corium bypass-delay walls of Zirconia. The material selection is in line with international experience gained from MCCI and Core Catchers experimental analysis.

The underlying concept of Corium-Barriers is that they are not required to remain in place in the very long term, but only as long as the containment would eventually fail by other means (e.g. overpressure due to slow pressurization). They are intended to delay containment bypass, which would eventually occur in the present design, so they should not be compared to Core Catchers in modern PWR designs especially when it comes to long term integrity.

The justification for this project is that it can be performed during the refurbishment outage, taking into consideration relevant matters as personnel dose, ease of transportation of material, required modifications in the installation for DBA and Normal Operation, time and cost.

Furthermore, it has been pointed out by the Operator that these types of improvements are in line with WENRA Requirements. WENRA Report Issue F "Design Extension of Existing Reactors" in its item F3.1 points out that DEC Analysis shall identify reasonably practicable provisions to prevent fuel damage (DEC A) and mitigate severe accidents (DEC B), in order to prevent large or early releases of to allow for sufficient time for prospective actions for the public to be implemented. It is also pointed out that confinement of radioactivity has the highest priority in DEC B. The Corium Barrier fulfills the objective of delaying large or early releases and extending the period in which confinement of radioactive material is ensured, compared with the original Plant design.

As the concept engineering of this strategy involves highly complex studies, it was proposed to incorporate it into the life extension program. The ARN has accepted this design improvement.

CNA II - Corium-Barrier to improve Containment long term integrity

The same principle of possible stabilization of molten material inside sump that has been applied for CNA I has been proposed for CNA II. The objective is to avoid containment breach due to Containment bypass through SIP suction lines, or at least delay to it, so as to decrease consequences in public as far as reasonable achievable.

CNA I and CNA II reactor cavity and sump design are not equal, and therefore each Plant requires a specific design for the corium barriers. Interferences in the sump are much more difficult, and the calculated volume and decay heat for corium is larger. Furthermore, possible solutions must not influence negatively Plant behaviour during DBA (mainly water accumulation in the sump during LOCA scenarios). Therefore, sump modifications must be carefully assessed.

The project in CNA II is in the conceptual design stage.

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 conducting the CAREM 25 Prototype 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.

ARN is committed to strengthen its licensing activities by actively participating in workshops, consultancy meetings and different forums for exchanging experiences, knowledge and lessons learnt. In the period covered by this National Report, ARN has participated in the IAEA First Consultancy Meeting on Small Modular Reactors Licensing (held in Ottawa-Canada). The objective of this meeting was to collect and document existing experience gained by regulatory bodies on the regulation of SMRs, including licensing and compliance assurance, with particular focus on challenges encountered, their resolutions, and insights into future issues.

The deliverable from this and the subsequent meetings is an IAEA TECDOC - "Practical Experience in the Regulation of SMRs" collecting the experience from abroad regulatory bodies such as Canada, the United Kingdom, Russia, South Africa, and China.

The TECDOC elaborates topics like the key regulatory challenges and lessons learned that have emerged in the regulatory decision-making related to SMRs in each Member States. For example, if there were changes in the legal and regulatory framework, changes in the design requirements, and safety analysis to face the licensing of an SMR. In addition, an attempt was made to gather information on other regulatory challenges, such as the requirements related to Physical Security and Safeguards, the Emergency Plan, etc. The TECDOC is currently in the final stages of revision.

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

The following activities have been completed for the implementation of the CNE External Emergency Control Center, called Municipal Emergency Operative Center (COEM):

- *Different sites around Embalse were analysed, and Almafuerte City, located 15 km away from the NPP, was selected as the most suitable place to install the CNE External Emergency Control Centre due to access suitability, availability of basic services and natural conditions.*
- *The temporary Municipal Emergency Control Center was set up in Almafuerte Firefighter´s Station through an agreement subscribed between NA-SA and Almafuerte Firefighter´s Department. The Center has been fully upgraded with technological equipments and the required furniture, identifications vests, computers, video systems, TVs, motor generators and an adequate communication room.*

The construction of a definitive COEM building is planned in a future step. The COEM building design has been completed taking into account ARN requirements and WANO and IAEA recommendations. NA-SA is in the process of searching a suitable place to allocate the building.

2.6. CNA I LONG TERM OPERATION 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 authorize 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 the relevant 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.*

ARN reviewed and approved all the submissions and renewed the Operating License 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 aimed at increasing the safety level of the plant. One of the main requisite is the comparison of the plant design against the current KTAs safety standards, as well as the latest German design requirements: RSH 3-01 "Fundamental Safety Objectives for NPP".

From this comparison a set of improvement / upgrade measures were identified by NA-SA leading to plant modifications. 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.*

In order to globally assess the cumulative impact of the proposed plant modifications, NA-SA developed a methodology based on the Periodic Safety Review's Global Assessment as per IAEA SSG-25.

Using the results of the above mentioned, NA-SA submitted to ARN in March 2022 the Integrated Implementation Plan with the final list of plant modifications necessary to face the Phase B and a proposal of schedule for timely implementation for all of them.

ARN reviewed and accepted the Integrated Implementation Plan based on the contribution of all the proposed improvements/upgrades measures to strengthen the defence in depth concept as stated in IAEA SSR2/1 Rev.1 and thus enhancing, as far as reasonably practicable, plant safety.

In addition, ARN has also considered in its decision making the remaining overall plant risk, as derived from the Integrated Implementation Plan.

Last July 1st 2022, the Board of Directors of NA-SA and ARN have signed the "Licensing Basis Document for Phase B" in which it is reflected the agreement between both institutions on high level licensing framework an activities for ensuring a safe LTO.

2.7. 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.

2.8. 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 NPP will be a HPR-1000 type reactor (Hualong I-1000), with a gross power of 1200 MWe, taking as a reference desing Fuqing unit 5 and including design changes according relevant updates of Argentinian AR 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 facing 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.*
- *To identify TSOs for new design features of HPR-1000, providing two types of support: participation in ARN reviews and, conducting independent assessment, analyses or experiments.*

According to the project schedule, it is expected that formal licensing activities will initiate at the beginning of 2024.

2.9. INDEPENDENT NUCLEAR OVERSIGHT

In October 2017, the Licensee of CNA I, CNA II and CNE implemented an Independent Oversight process based on WANO / IAEA guidelines.

The objective of the Independent Oversight is to promote excellence in the operation of nuclear power plants throughout the company and to provide the CNO, plant managers, 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 predefined annual and long-term schedules, 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 Independent Oversight.

2.10. ATUCHA SPENT FUEL DRY STORAGE

Based on the spent fuel pool capacity and planned operation of CNA II NPP, NA-SA has decided to face the construction of a Spent Fuel Dry Storage.

In a similar way to the CNA I's spent fuel dry storage, the realization of the project was assigned to the Argentinian Atomic Energy Commission (CNEA). The technical specifications are being reviewed. This facility is scheduled to start operation in March 2026.

The CNA I spent fuel dry storage (ASECQ) is going to be operative in 2022. This facility contemplates not only finishing Phase A, but also operation of CNA I for about 3.5 full power years in Phase B of the Long Term Operation.

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. *In 2019, CNE began its second operating cycle for a period of 30 years, while CNA I has been operating at stage A of the Long Term Operation since 2018.* Besides, the CAREM prototype reactor (CAREM), *a domestically-designed small modular reactor*, 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 (gross) and began its commercial operation in 1974. Nowadays, the gross power output of CNA I is 362 MW. It is a 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, about 110 km to the south of the homonymous city, with 600 MW(e) (net) began its commercial operation in 1984. It is a PHWR reactor of CANDU type, natural uranium loaded and heavy water moderated and cooled. *After the CNE's refurbishment outage, the power output of CNE increased by 8 MW.*

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:

- Operating Experience Management Program
- Ageing Management Program including plant programs
- Training and qualification of operating personnel
- Quality Assurance Program
- Severe Accident Management Program

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

3.6.3.1.1. 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.

3.6.3.1.2. Ageing Management Programs and activities

Ageing Management Programs are one of the mandatory documentation for granting a License and keeping valid the licensing basis. The program was implemented to ensure that the degradation mechanisms and the ageing effects do not affect the capability of the SSCs to carry on their planned safety function, throughout the whole life cycle, taking into account the changes that occur in these SSCs due to the time and usage.

Argentina follows the IAEA Safety Report Guidelines (SSG-48 and SRS No. 82) to develop the Ageing Management Programs in its NPPs. The ARN's expectations are aligned with this standard for an effective implementation of the ageing program and 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 took 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, and so to assess whether all ageing effects and degradation mechanisms are being managed and will continue be managed through the extended period of operation.

CNA I performed, as part of ageing management review, a total of 83 condition assessment reports corresponding to 60 systems (SC1, SC2 and some SC3 systems) for continued operation after the end of life as defined by the original designer. The results of these 83 reports identified 576 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 and some others were related to replacement or refurbishment.

CNE is working on developing the ageing management review table, in accordance with the aforementioned IAEA guidelines.

In 2019, CNE issued a high level document called "Ageing Manual" as a mandatory documentation for the licensing process. This document stablishes the framework to develop the different Ageing Management Programs based on IAEA - IGALL. As part of the development, procedures to set the scope of the SSCs were issued and scope reports have been created. This systematic process is ongoing with priority on safety systems and safety related systems.

CNE successfully re-started its normal operation in August 2019, after the refurbishment outage. Since the beginning of this period, 988 recommendations, related to the condition assessment (114) and life assessment (22) reports, has been taken into account. Periodically, ARN monitors their implementation.

In addition, CNA I, CNA II and CNE are improving their current Ageing Management Programs 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 already exists 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. These continuous improvement activities are benefit by the participation of experts from NA-SA - CNA I/CNA II/CNE and from ARN in the different working groups of the IGALL meetings (International Generic Ageing Lesson Learned), organized by IAEA, in order to learn from the experienced NPPs, contribute to the development and update of the generic AMP's and evaluate their applicability in each particular case.

In Argentina it is also required to manage non-physical ageing of SSCs as their process of becoming out of date. This regulatory requirement gives an essential role to the Obsolescence of SSCs Program and, so any associated decrease in reliability and availability can be resolved with foresight and anticipation.

All the three NPPs, CNA I, CNA II and CNE, have recently updated the Obsolescence Program using as a main reference document the IGALL, TOP401 "Technological obsolescence Program" – IAEA.

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 2022* 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 pool building.

The ASECQ will become a new building attached to and integrated into the existing building of the spent fuel pool building (hereinafter CP1). The building will include a silo located below the 0.50 m level, which will house the 2,844 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 back 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 pool 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 pool 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 4, 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 uninhabitability of the main control room due to toxic and / or corrosive gases, release of radioactive materials or sabotage. An Instruction for operation and monitoring of the plant from the emergency control room was developed.

3.6.4.1.3. *Review and improvement of emergency procedures*

- *Instruction T-42 "Unavailability of the Main Control Room" will be incorporated into the Operating Manual, which will enable scram to be made from the Emergency Control Room located in the Second Heat Sink building, in order to bring the plant into a safe condition, until the Main Control Room can be recovered. This instruction is in process of approval.*
- *New level and temperature measurements were 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. They are included in the guideline 1-GAS-CE 12 - Water Injection to the Spent Fuel Elements Pools.*
- *Instruction T-18: "River Drought": the criteria related to the water level in the Paraná River was reviewed and additional manoeuvres and monitoring actions were introduced.*

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 (ORE), they changed from Severe Accident Management Guidelines to internal instructions of the ORE. 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.

During the period of this report, the following improvements were implemented:

- *New level and temperature measurements were 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. They are included in the guideline 1-GAS-CE 11 - Water Injection to the Spent Fuel Elements Pools.*
- *Internal instructions were drawn up within the framework of severe accident management:*
    - o *ORE-015: Water supply to the tanks of the GHC system - Demineralized water from the SGA fire system- Fire network.*
    - o *ORE-007: Power supply SGA system from the fire network of the UG-PN workshop.*
- *River drought and flood: the criteria related to the water level in the Paraná River was reviewed and additional manoeuvres and monitoring actions were introduced.*

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:

- *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).
- *2-GAS-CE-05: "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 and 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.

3.6.4.2.4. New design of the Upper Guide of Neutron Flux Probes

During the reactor internals inspection carried out during a planned outage, buckling in the guide tubes of two neutron flux probes was observed. The upper part of the guide tube, called upper guide, got stuck during the transition from hot shutdown to cold shutdown. When the reactor cooled down, there was a clogging between the upper guide and their respective housings in the tank lid of the moderator. When contracting and not being able to slip into the hole, the guide tubes buckled.

The causes of the jamming between the upper guide and the orifice of the moderator tank lid were related to the design and assembly of the guide tube.

To address this issue, NA-SA began working on developing new designs for the upper guide. As a result, five new designs were studied. A Mock-Up facility was designed and built to simulate the movements of the moderator tank generated during different load states and operating transients that can affect the functionality of the upper guide.

After testing in the Mock-Up and evaluating all the upper guides studied, one of them (identified as UG5) was chosen. No friction force was observed when using the UG5 upper guide during the traction-compression cycles similar to that is generated in the reactor for different degrees of misalignment, whose main features are as follows:

- *Friction force generated is acceptable and does not cause buckling load.*
- *Misalignments do not cause an increase in the maximum force.*
- *No interaction between springs and the outer piece that contains them was observed.*
- *A suitable contact was verified for various misalignments between surfaces of the moderator tank lid and the upper guide.*
- *The misaligned assembly of the upper guide was tested, verifying that it tends to align itself when touching the lower cone.*

3.6.4.3. CNE IMPROVEMENTS

3.6.4.3.1. Safety Improvements in the frame of Life Extension Project

Numerous improvements were implemented during the period *2016-2019* related to the CNE life extension project. Those related to *design changes and improvements in safety systems and related to safety systems are mentioned below:*

- *Reactor re-tubing: removal and fuel channels replacement, including pressure tubes, calandria tubes, end-fittings, feeders and others components like calandria tube inserts.*
- *Steam Generators (SG) replacement: involves replacement of the cartridges (SG´s tube plate and tube´s bundle), pressure vessel envelope, and the steam drum internals (only primary moisture separators).*
- *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.*
- *Emergency Core Cooling System (ECCS) reliability improvement: includes the automatic initiation for smaller LOCA events, automatic switch from medium to low pressure injection, relocation of sump level sensors to improve measurement, tripling of the level measurement of the dousing tank, Primary Heat Transport System (PHTS) sustained low pressure automatic initiation of low pressure injection, etc.*
- *Reduce interface LOCA event frequency through the ECCS line.*
- *Automatic trips to protect main PHTS pumps.*

- *Improvement of seismic capacity of Emergency Power Supply and Emergency Water Supply systems.*
- *Improvement of the response to severe accident, by several design provisions: Passive Autocatalytic Recombines to remove hydrogen from the containment and promote mixing within Reactor Building; installation of Filtered Containment Venting System to prevent loss of containment structural integrity as a result of over-pressurization, addition of external feed water inside the calandria vault and a rupture disc.*
- *Replacement of the Digital Control Computers.*
- *Replacement of the Diesel Generators Class III.*
- *Replacement of the Moderator Heat Exchanger´s new design.*
- *Repowering and addition of a fifth pre heater in the Turbine.*
- *Implementation of improvements derived from condition and life assessments.*

Also activities were carried out related to updating of safety studies such as Probability Safety Assessment 1 and 2 (which include design changes and new hypothesis), and Deterministic Safety Analysis. A severe accident program and equipment environmental qualification program were implemented, as well.

A particular scheme of return to service was developed by ARN in order to allow the systems and components to enter into normal operating phase (assembly and conditioning phase):

- *Systems that were not modified or intervened during the Refurbishment outage were returned to service, taking them from the state or configuration in which they were during the outage to normal operation state according to the existing operational documentation.*
- *For intervened or installed systems and components, its return to service was consisted in 2 (two) types of tests defined as:*
    - *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 according to the new requirements.*

The Commissioning Program was prepared by the licensee and submitted to the ARN for approval and covered 3 (three) phases: A, B and C.

ARN established several milestones in order to continue to the next phase and defined pre requisites to be met to overcome them and continue with the commissioning process.

The phases of the Commissioning Program were defined as:

- *Phase A: covered from the end of assembly and conditioning of the Structures, Systems and Components to the loading of fuel elements. A successful execution of the hydrostatic test of the Primary Heat Transport System was defined as milestone for the end of this phase.*
- *Phase B: covered from the complete loading of fuel elements to removal of guaranteed shut down state up to 5% full power. A successful Containment Test was required as a milestone to remove the guaranteed shut down; and the constitution of an ad-hoc Committee from the Licensee composed of qualified persons with experience in the design, construction, start-up and operation of nuclear power reactors to evaluate the mandatory documentation required for the Commissioning before its submission to ARN was required as well.*
- *Phase C: covered the stage of power increase and was divided in 3 (three) steps of 50%, 80% until reach 100% of Full Power. The successful execution of the tests corresponding to each power step constituted a milestone to proceed with the authorization to the next step.*

Once the commissioning tests were completed, the ARN has approved the Final Safety Analysis Report issued by the ad hoc start up committee and other mandatory documentation required for the Second Cycle of Operation of CNE, and has granted the Operation License on August 22nd, 2019.

3.6.4.3.2. Safety Improvements during the reported period

3.6.4.3.2.1. Alternative heat sinks

- *The cold water system capacity increased by 100%. This increase allows the system to feed containment LAC (Local Air Cooler) 1 to 8.*
- *Redundant high pressure nitrogen supply to the containment spray system valve actuators.*
- *A bypass line to an ECC (Emergency Core Cooling) heat exchanger was installed.*
- *Facility for the pneumatic remote opening of the main steam safety valves.*

3.6.4.3.2.2. Instrumentation and control (I&C)

- *Regarding obsolescence issues, the whole radiation activity measurement chains of the containment isolation system were replaced. The new detectors have Environmental Qualification.*
- *Main generator synchrony meter was replaced.*
- *The drying tower of the pressurization system of the high-pressure ECC tank was replaced.*

3.6.4.3.2.3. Processes

- *The pump motor of the high-pressure service water system was replaced, allowing the system to have a higher capacity.*
- *Regarding obsolescence issues, the emergency diesel-driven air compressor was replaced.*
- *Regarding cavitation issue, the moderator temperature control valves were replaced with a new one that has a custom anti-cavitation systems.*

3.6.5. IMPACT OF THE PANDEMIC

The management of all activities mentioned in this report was carried out in a safety way. Since the COVID pandemic began, the whole operation organization carries out its activities under the Hygiene and Safety Protocol following the general and specific preventive measures for Nuclear Power Plants with the objective of guarding the hygiene conditions of the facilities both to prevent infection and to continue normal operation of the NPPs.

As specific measures in the sites, it stands out the implementation of a reorganization of minimal shifts of operation along with sanitary protection for 14 days, a strict control over the workplace hygiene, and the adoption of exclusive means of transportation for the operation personnel.

Furthermore, in the case of planned and unplanned outages, a guide of complementary recommendations was established to be followed during planning and execution of them. This allowed schedules to be followed beyond some particular delays in the start date.

With regards to all inherent activities within the scope of the current Nuclear Safety Convention, it is worth pointing out that those all activities were carried out through the implementation of minimal shifts of operation established in the Protocol, guaranteeing the appropriate safety standard.

In addition, all measures established in the aforementioned Protocol were constantly updated according to the state of the health situation, gradually resuming the pre-pandemic scheme and keeping the company's preventive measures.

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 by 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.
- *Resolution ARN No. 333/2019 by which ARN granted NA-SA the Operating License for CNE so this NPP has returned to service for another 30 years of operation following completion of upgrade work.*

  The Resolution of the ARN Board of Directors No. 477/16 authorized the Responsible Entity to intervene in the systems of the installation with the reactor in the state of Guaranteed Shutdown, in order to execute the tasks related to the Life Extension of the Embalse Nuclear Power Plant. Likewise, it was determined that said Operating License would be in force until the removal of the guaranteed state of shutdown is necessary.

  In this context, NA-SA completed the activities and tests of the Commissioning Program, and the mandatory documentation complying with the regulatory requirements applicable to the installation. Likewise, on August 1st, 2019, NA-SA requested the Operating License for the Second Life Cycle of the CNE.
- *Resolution ARN No. 173/2019 approved the Draft Standard "Management System for Safety in Facilities and Practices" Rev. 0, through which it declared the opening of the Participatory Standard-Making procedure, allowing the participation of stakeholders. Resolution ARN No. 036/2020 approved AR 10.6.1. "Management System for Safety in Facilities and Practices" and Resolution ARN No. 397/2020 established the entry into force as of April 1st, 2021.*
- *Resolution ARN No. 185/2019 approved the Draft Standard AR 10.1.1., Basic Standard on Radiation Safety Rev. 4, which it declared the opening of the Participatory Standard-Making procedure, allowing the participation of stakeholders, and Resolution ARN No. 521/2019 approved the final text of AR 10.1.1. Rev. 4.*

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, which was a body belonging to the regulatory branch of CNEA in that period).

With the time, a normative system was established comprising areas such as radiological and nuclear safety, safeguards of nuclear materials and physical protection and security of radioactive sources. The system, known as "AR Standards", at present consists of 64 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.

There is a Regulatory Norm Division, depending from the Board of Directors, with the mission to elaborate, review and update the regulatory Standards and Guides.

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
3.7.1.	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
10.6.1.	*Management System for Safety of Installations and Practices*
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	*Dosimetry Factors for the external and internal exposition, guidance levels of radionuclides in food and water and recommendations for the exposition control to radon gas.*
AR 3	Conditions to be verified by the specialized physician according to the phychophysics for the specified function
AR 8	Generic clearance levels
AR 10	Specialized training program and specific training for licensing of personnel of Type I radioactive installations
AR 13	Storage of radioactive waste
AR 14	Design and Development of a Radiological Environmental Monitoring Plan

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 the Argentinean Regulatory Standards with the IAEA Safety Standards. As result of such harmonization, it was concluded that Argentine Regulatory Standards are aligned 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 meetings to take into account the lessons learned from the Fukushima accident. This is reflected in the national normative review, considering the updated adjustments of IAEA Standards when assessing the Argentinean normative, in order to strengthen the nuclear safety and achieve the IAEA Action Plan objectives for 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 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 to review the national normative framework.

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 the Vienna Declaration. This review would include the updating of the standards in force and the development of new ones, when necessary.
- Harmonize ARN Regulatory Standards 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 the preparation for the next Integrated Regulatory Review Service (IRRS), that will be carried out in Argentina.

The 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 includes the elaboration of 18 new standards and 7 new guides; and the revision of 11 standards and 1 guide.

During the period 2019-2022, it was reviewed the AR 10.1.1 "Basic Radiation Safety Standard". This updating aroused the need to update the Regulatory Guide AR 1 "Dosimetry Factors for the external and internal exposition, guidance levels of radionuclides in food and water and recommendations for the exposition control to radon gas", applicable to type I installations and updated to facilitate the Regulatory Standard AR 10.1.1 fulfilment.

Also, it was issued the first edition of the standard AR 10.6.1."Management System for Safety of Installations and Practices": This new standard takes the regulatory verification activities from the management system for safety of the Responsible Entities. This Standard is harmonized with international consensus and the IAEA GSR part 2.

Besides; there was a substantial evolution in a Standard related to the Licensing of NPPs. This is a new specific standard, in development, in the Normative Framework.

Other new standards applicable to NPPs, which are under development are:

- Glossary, that incorporates new terms and harmonize their use in Standard Corpus.
- Preparedness and Response for a Nuclear or Radiological Emergency, that takes into account recommendations of IAEA Safety Standard No. GSR Part 7.
- Safety requirements for nuclear reactors construction, which is foreseen to contain 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, supposed to incorporate the requirements for the licensees to complete these documents.
- Periodic Safety Review for NPP that contains the requirements issued for Periodic Safety Review (PSR) of CNE and CNA I.
- Safety requirements for the design of NPP, this new standard is expected to contain 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 ten 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) 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) 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) 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) 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.

In 2019-2020 the technical areas and the legal department of ARN worked jointly on the revision and updating of the investigation procedure for enforcement measures (Resolution ARN No. 75/99).

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 by 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 Response System (SIEN in its Spanish acronym), complemented by the SIER (Radiological Emergency Response 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.

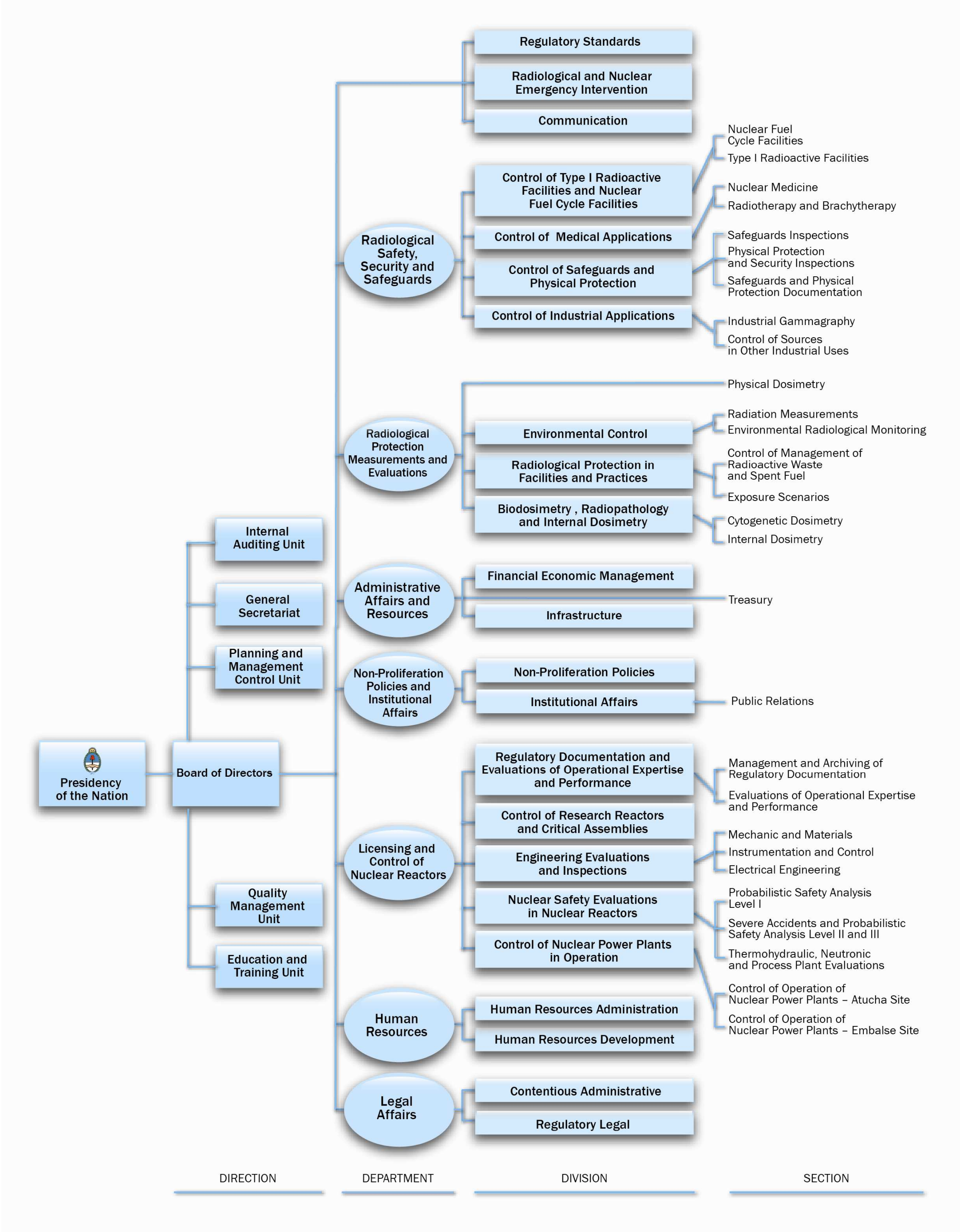


Figure 3.8.1. – ARN Organization Chart

The General Secretariat, as well as the Internal Auditing, Planning and Management Control, Quality Management and Education and Training Units (abbreviated to UCE in Spanish), report 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 it is also the manager of some common infrastructure sectors as Information Technology and Information Registry.

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 tasks fulfilment of the approved annual working plan and 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 carries out regulatory assessments and inspections concerning radioactive installations (medical, research and industrial installations), nuclear fuel cycle installations, transport of radioactive and nuclear materials, safeguards control, security of radioactive sources 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, *radionuclides determinations, biological dosimetry, internal dosimetry, environmental monitoring* 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 *siting*, design, construction, commissioning, operation and *decommissioning* 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 *enforcement process*, purchasing processes, contracting and licensing.

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

The Regulatory Standards Division is responsible for *developing and revising* the ARN 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 developing the ARN functions related to radiological and nuclear emergencies preparedness, training and response (see article 3.16.).

The Communication Division 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 363 personnel. Nowadays, 59% of 363 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 57% of ARN personnel have university degree, either master degree or Ph.D.

The distribution of professional staff is the following: 38% belongs to engineering branch, 28% to Natural Sciences area (physics, chemistry, biochemistry, environmental and medicine), 6% is specialized in social sciences, 11% belong to economic sciences, 2% to legal area and the remaining 15% belongs to other specialities.

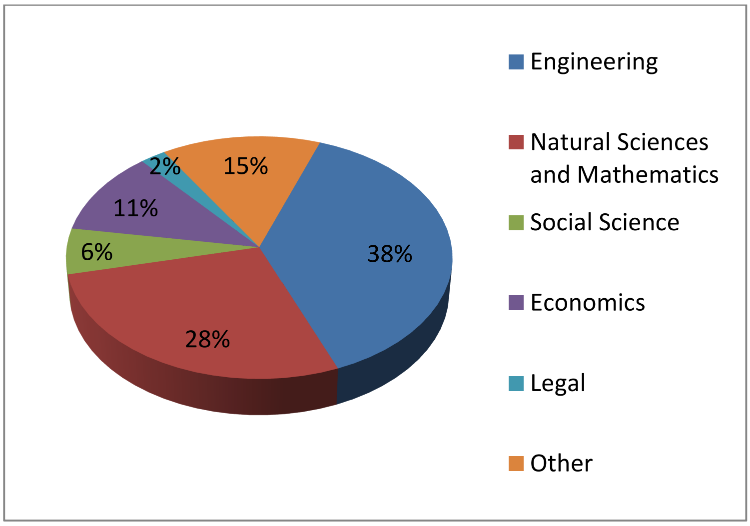


Figure 3.8.2. – ARN professional staff distribution

In relation to engineering branch the ARN is composed of 23% Electronic, 6% Electric, 4% Nuclear Physicians, 32% Chemical, 2% Civil, 10% Industrial, 11% Mechanical and 12% to other specialities.

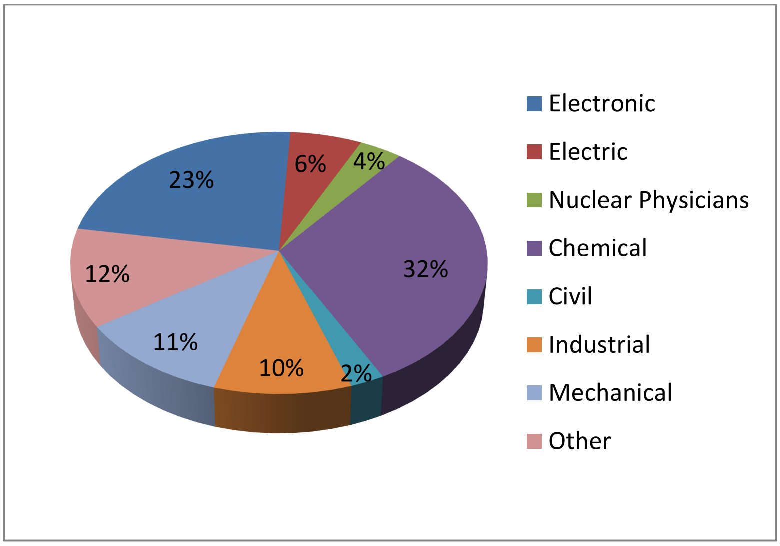


Figure 3.8.3. – Distribution based on engineering branches

The Natural Sciences area is composed of 27% physics, 22% chemistry, 17% biochemistry and biologic science, 9% environmental, 3% medicine and the remaining 22% belongs to other specialities.

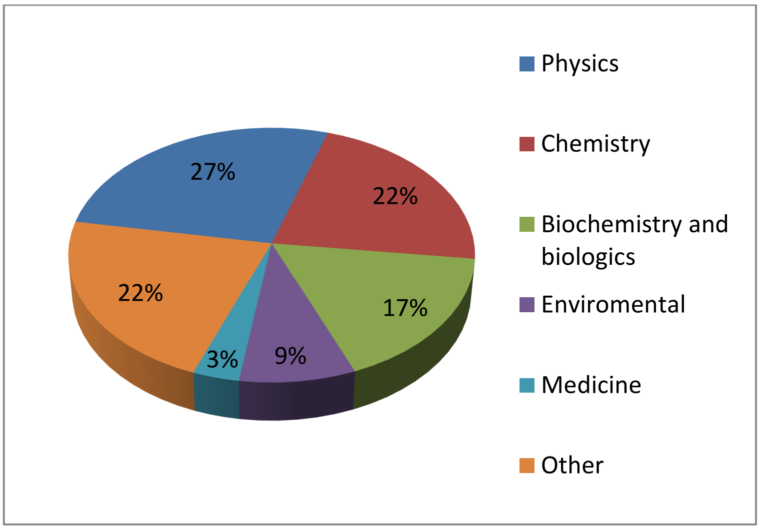


Figure 3.8.4. – Distribution based on natural sciences disciplines

ARN´s staff is geographically distributed as follows: 75% at headquarter in Buenos Aires City, 21% in Ezeiza Atomic Centre, 3% in Nuclear Power Plants and 1% in Ushuaia and Bariloche (CTBTO monitoring facilities).

In addition, the Board of Directors of ARN counts on 10 advisers´s collaboration, related to different regulatory matters.

Finally, there are 7 ARN´s experts temporarily performing functions in international organizations in the following way: 5 persons in the Brazilian-Argentine Agency for Accounting and Control of Nuclear Material (ABACC) and 2 in the IAEA (International Atomic Energy Agency).

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 to participate 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).

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.

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 (See section 3.8.3.1.5.).

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.

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.).

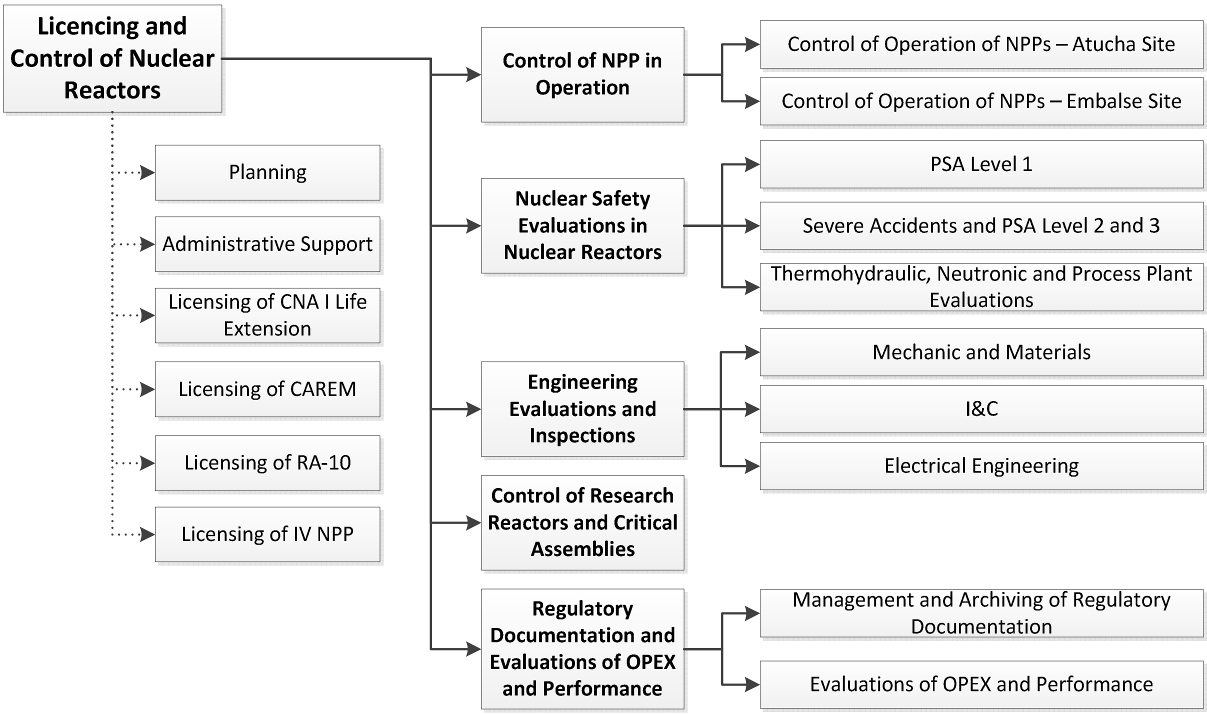


Figure 3.8.5. – Structure of the Licensing and Control of Nuclear Reactors Department

About *60* persons 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 *2022* is the following:

Inspections and evaluations in NPPs	*82%*
Support activities directly related to safety	*13%*
Support activities indirectly related to safety	*5%*

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.

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. *However, due to the pandemic situation, no audits were carried out during the reported period.*

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.

The current activities for the regulatory control of CAREM's construction require the deployment of inspectors in the civil area. This is done with the assistance of TSO.

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. A summary of activities in this regard is presented below.

Since 1980, every year the ARN has taught two Postgraduate Training Courses in Radiological and Nuclear Safety with the collaboration of professors from the Engineering School of the University of Buenos Aires on nuclear physics. Students from Latin America, the Caribbean and even from other countries of the world have participated in them. With the same frequency and in the same period, the ARN dictates a Basic Training Course in Radiological Protection for undergraduates.

In 2013 and 2014, two important events occurred that should be highlighted. The University of Buenos Aires granted the ARN Postgraduate Courses in Nuclear Radiological Safety the status of Specialization Career in Radiological Protection and Safety of Radiation Sources and Specialization Career in Nuclear Safety respectively, which in turn were accredited by the National Commission for University Accreditation (CONEAU).

Also noteworthy are the EduTA Missions to Argentina. 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.

As in the rest of human activities, the COVID-19 pandemic caused a significant disturbance in the educational ARN activity. So much so that in 2020, a few weeks after its start, the Specialization Degree in Radiological Protection had to be suspended and foreign students returned to their countries. In the same year, the Specialization Career in Nuclear Safety was also canceled and for the first time the Basic Training Course on Radiological Protection was taught entirely online.

With the pandemic still ongoing, in 2021 all the theoretical classes of the Specialization Career in Radiological Protection were delivered in a virtual mode and the practical activities and study visits were postponed to 2022. The number of attendees was 29; from which 17 professionals were from Latin America and the Caribbean region’s countries. Regarding the Specialization Career in Nuclear Safety, its realization was suspended due to the pandemic and the Basic Training Course on Radiological Protection began to have the formal sponsorship of the IAEA, which awarded scholarships to foreign students. Three members of the ARN staff participated in it.

The edition of the Specialization Degree in Radiological Protection corresponding to 2022 is being developed combining stages of virtual and face-to-face dictation (blended learning). The number of attendees is 28, form which 4 are from the ARN staff, 10 from other Argentine organizations and 14 from Latin American and Caribbean region’s countries. In addition, in 2022 the practical activities and scientific visits of the 2021 Edition of Specialization Degree in Radiological Protection will be performed, the teaching of the Specialization Degree in Nuclear Safety will be restarted in a blended learning mode and the Basic Training Course on Radiological Protection will be taught virtually.

During the period 2019 to 2021, 2 ARN professionals achieved the Specialization Degree in Radiological Protection and 2 ARN professionals the Specialization Degree in Nuclear Safety. Additionally, 28 professionals from Latin America and the Caribe got the Specialization Degree in Radiological Protection and 16 the Specialization Degree in Nuclear Safety. During the same period, 61 under graduate workers attended to the “Basic Training Course on Radiological Protection”, 5 of them were ARN staff.

Other activities managed by the UCE, related to the strengthening of the capacities of ARN personnel and other interested parties, were:

- The organization of courses on specific topics requested by different Divisions and Departments of the ARN.
- The organization of tailor-made courses required by different interested parties.
- Active participation in the PGEC Steering Committee of the PGEC, Directors meeting, and
- ARN experts also participate in safety-related activities in organizations such as ICRP, UNSCEAR and the IAEA (eg CSS, INSAG, IRS, NSGC, NUSSC, RASSC, TRANSSC, WASSC) as consultants and also attend technical meetings. They also participate in OSART, IRRS and international expert missions. Each activity carried out allows for a fruitful exchange of experiences and lessons learned, and promotes a high level of competence in safety.

3.8.4. QUALITY MANAGEMENT SYSTEM IN THE REGULATORY BODY

The ARN implements its quality management system to continuously improve the effectiveness and efficiency of its regulatory actions, focused on increasing the satisfaction of the interested parties, considering that the requirements on environment, health, quality, occupational safety, information security, physical protection and economic aspects are integrated into the nuclear and radiological safety requirements.

To achieve compliance with these requirements and consequently the satisfaction of the interested parties, the ARN:

- *Identifies the processes.*
- *Determines their sequence and interaction.*
- *Determines the criteria and methods necessary to ensure that the management of each process is effective.*
- *Ensures the availability of resources and information necessary to support the operation and monitoring of these processes.*
- *Performs the monitoring, measurement and analysis of these processes.*
- *Implements the necessary actions to achieve the planned results and continuous improvement of these processes.*

The QMS requirements are described in the Quality Management Manual (MC-ARN), which was updated in 2022.

The ARN Board of Directors declares and communicates the Quality Policy.

QUALITY POLICY

The ARN assumes the commitment to protect people and the environment, in the present and the future, from the harmful effects of ionizing radiation and to control that regulated activities are carried out for exclusively peaceful purposes.

ARN promotes a safety culture based on a questioning attitude, on a rigorous and prudent approach in regulatory actions, and on transparency in access to information on radiological and nuclear safety aspects for stakeholders.

Based on its strategic goals, the ARN plans, carries out, verifies and acts for the continuous improvement of its Quality Management System, based on the requirements of the ISO-9001 Standard in its current version and considering the international recommendations to implement a Integrated Management System.

3.8.4.1. PROCESS APPROACH IMPROVEMENT

The ARN has a Process Map, which reflects the necessary processes and their interactions to comply with its regulatory functions.

Process management is described in the mandatory document called the Process Table.

The ARN maintains the corresponding records to support the operation of its processes and to have confidence that they are carried out as planned in favour of safety.

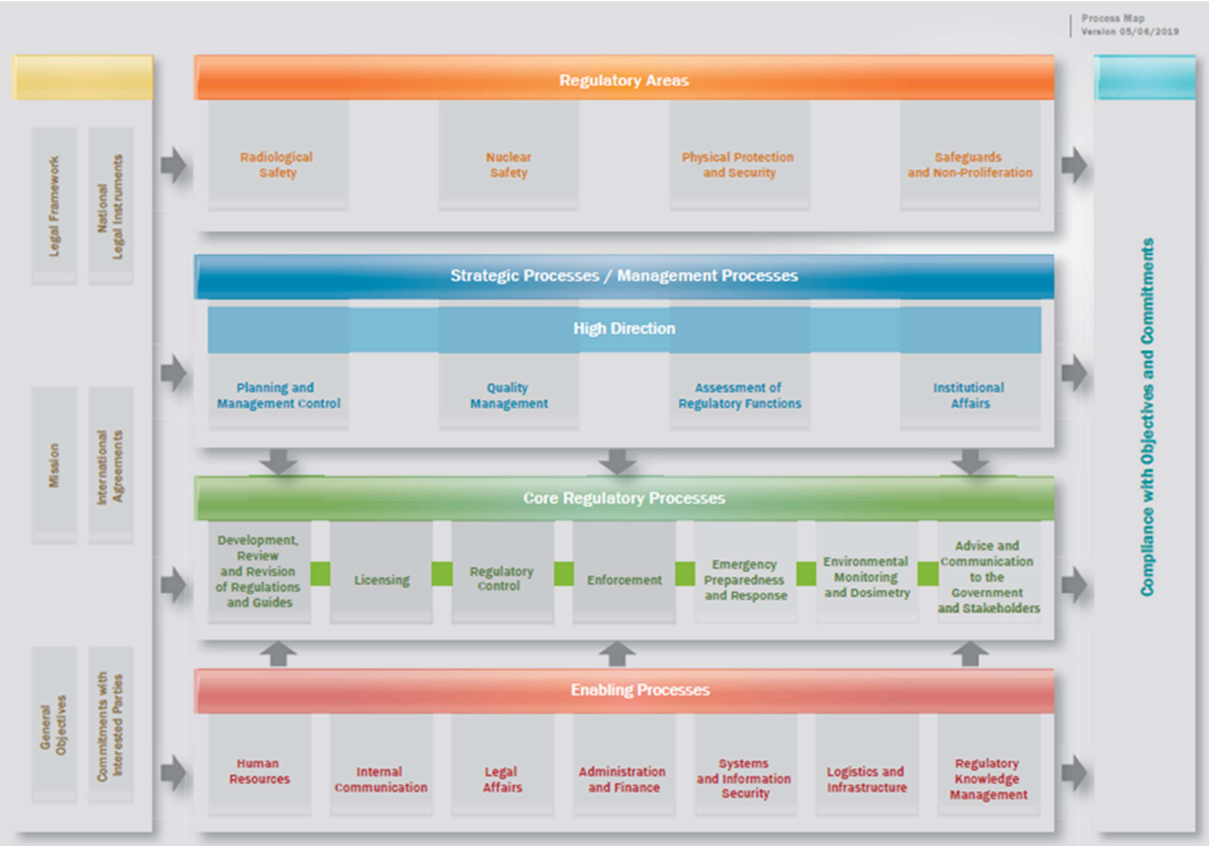


Figure 3.8.6. – ARN Process Map

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.

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.

The assessment of the quality management performance and effectiveness is done through different methods:

- *Monitoring of processes.*
- *Evaluation of indicators of fulfilment of objectives in the processes.*
- *Survey analysis.*
- *Internal audits and quality checks.*
- *Management review.*
- *Control by oversight.*

Internal quality audits verify the implementation and effectiveness of quality management. They are carried out by independent auditors of the process being audited, who have the appropriate qualification. The Quality Management Unit proposes an annual program of internal quality audits to Senior Management for evaluation, review and approval.

During the period covered by this report, the Quality Management Unit has carried out 23 internal audits and 2 external audits of the COVID-19 Protocol carried out by the Argentine Institute for Standardization and Certification (IRAM) and 7 external audits of Laboratories Accredited by the ISO-17025 standard carried out by the Argentine Accreditation Agency (OAA).

The review by the High Direction, the performance of the Quality Management System is assessed to ensure its suitability, adequacy, efficiency and effectiveness. As a result of such review, opportunities for improvement, change needs in the quality management system or resource needs can be identified.

3.8.4.3. CERTIFICATION AND ACCREDITATION

Since 2020, the ARN has a COVID-19 Protocol certified according to the IRAM-3820 Standard by the Argentine Institute for Standardization and Certification (IRAM).

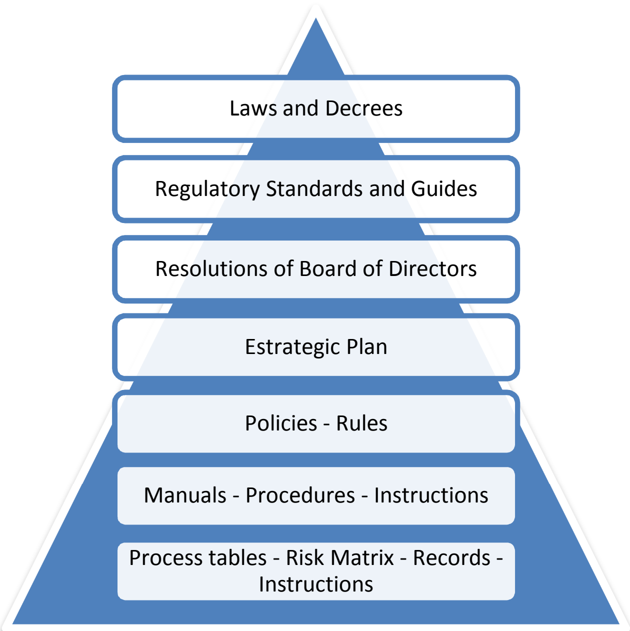
Likewise, since 2006, the Radiological Protection Measurements and Evaluations Department has accredited laboratories and is implementing accreditation under the ISO/IEC Standard (17025:2017) "General requirements for the competence of testing and calibration laboratories" and the established criteria by the Argentine Accreditation Body (OAA).

To date, the accredited laboratories are: the Multisite Testing Laboratory belonging to the Biological Dosimetry Laboratories and the Thermoluminescence Dosimetry Laboratory satisfactorily completed the evaluations for the first maintenance to full standard (2020-2021 period), as well as the second full standard maintenance to the Environmental Control Laboratory corresponding to the 2022 period.

Likewise, the Calibration Laboratory (LDF-LC 029) satisfactorily carried out the evaluation of the 1st and 2nd maintenance to full standard (period 2021-2022)

3.8.4.4. DOCUMENTATION REQUIREMENTS

The documental structure of the QMS, is made up of external and internal documentation that provides a legal framework for the development of the different activities of the ARN.



The organization keeps as documented information everything what is required by ISO:9001 in its current version and what is determined by the organization in its respective documents such as manuals, procedures, process tables, instructions, rules, among others.

Documentation is controlled, usable, legible, clearly identified and easily accessible at the point of use.

Likewise, ARN has a defined Information Security policy and during the year 2022 an Information Security Committee has been created in order to ensure the effectiveness of the management system processes and unify the criteria for strategy inclusion of the information security in all the institution's projects.

During the period of this report, 117 documents (manuals, rules, procedures and work instructions) have been issued or updated.

3.8.4.5. SATISFACTION OF STAKEHOLDERS

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

ARN ensures that the interested parties requisites are known by the organization personnel and satisfied as far as possible, giving the first priority to the fulfilment of the National Nuclear Law, Act No. 24,804, protection of the population and workers, as well as the environment. Likewise, appropriate strategies are defined for interaction with them:

- *Appropriate means to communicate regularly and effectively with the interested parties and inform them regarding the radiological risks associated with the operation of the facilities and the performance of the activities;*
- *Appropriate means to communicate in a timely and effective manner with interest parties in the face of changed or unforeseen circumstances;*
- *Appropriate means of disseminating necessary safety-related information to interested parties;*
- *Appropriate means to consider in the decision-making processes the concerns and expectations of the interested parties in relation to safety.*

3.8.4.6. CONTINUOUS IMPROVEMENT

To improve the efficiency and effectiveness of the quality management system and meet the requirements of the interested parties, the ARN analyzes the data from internal quality audits, compliance indicators, High Direction reviews, survey analysis, suggestions and opportunities for improvement from interested parties, implementing the following actions:

- *Identification of needs or opportunities that must be considered for continuous improvement. Likewise, the processes can correct, prevent or reduce undesired effects in order to contribute to the improvement of the quality management system.*
- *Implementation of appropriate corrective actions without undue delay for nonconformities and observations of the quality management system that are detected during an internal quality audit or that are self-detected.*

3.8.5. COMMUNICATIONS WITH THE STAKEHOLDERS

ARN has *both* the legal obligation to inform the public and the willingness to communicate with stakeholders. *ARN aims to communicate about radiation and nuclear safety in a transparent, timely and understandable manner.*

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.

The ARN's website is an important tool in communication, which provides information about 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.

During the COVID-19 pandemic, a special section was created with the news about the regulatory operation during COVID-19.

ARN also uses social media (Facebook, LinkedIn and YouTube) as two-way communication tool for interaction with public. 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 *2021-2025 places communications as one of the five strategic lines with the aim to consolidate a distinctive institutional image.*

Internal communication provides personnel with the information about ARN activities. The internal site for staff was redesigned, creating new sections, a space that integrated all the information regarding COVID-19 and the publication of news was streamlined. The creation of the YouTube channel made it possible to share internal videos with the staff, such as the protection measures for COVID-19, and also, technical conferences for knowledge management.

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 persons and institutions compelled to pay these regulatory fees.

The budget assigned to the ARN for the financial years 2019 and 2022 are shown in Table 3.8.1. An important increase of the 2022 budget on personnel can be observed. This allows ARN to achieve the strategic goals related to human resources development, in order to accomplish the licensing of the new projects mentioned before and continue with its normal tasks. The total budget during 2022 is composed of: 41% from the National Treasury, 57% from annual regulatory fees and goods or resources assigned according to applicable laws and regulations, and 2% from donations.

Table 3.8.1. – ARN comparison budget for financial years *2019 and 2022*

Item	*$ (in thousands of Argentine pesos)*	
	2019	***2022***
1 - Personnel	*435,927*	*793,351*
2 - Support goods	*15,309*	*20,840*
3 - Services	*137,052*	*262,989*
4 - Equipment	*21,357*	*45,817*
5.1 - Fellowships	*4,140*	*1,740*
5.9 - Transfers	*56,740*	*386,863*
6 - Others Financial assets	*129,701*	*249,033*
9 - Other expenses	*151,606*	*238,217*
TOTAL	***951,832***	***1,998,851***

The ARN percentage budget distribution is shown in Figure 3.8.7.

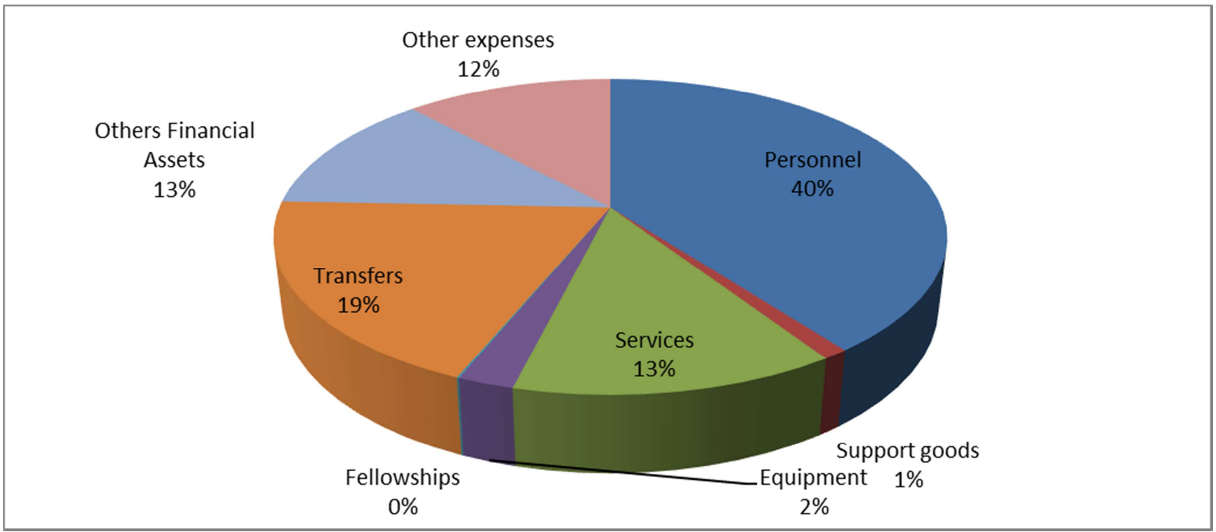


Figure 3.8.7. – ARN *2022* percentage budget distribution by category

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 2019 until now, in the following meetings:

EVENT	***PLACE***	***DATE***
Plenary *Technical Committee*	*Santiago de Chile, Chile*	*June/July, 2019*
Technical Committee	*Recife, Brazil*	*December, 2019*
Plenary *Technical Committee*	*Virtual*	*July, 2020*
Technical Committee	*Virtual*	*October, 2020*
Plenary *Technical Committee*	*Virtual*	*November, 2020*
Technical Committee	*Virtual*	*March, 2021*
Plenary *Technical Committee*	*Virtual*	*June/July, 2021*
Technical Committee	*Virtual*	*November, 2021*

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 Secretary of Civil Protection of the Nation and ARN. August, 2019.	*Establish a "National Platform for Alerts and Monitoring" in which the member organizations provide information of interest for the protection of the public and upload data to said digital platform to alert with a sufficient degree of certainty and timeliness about any imminent situation of risk.*
Agreement between the Nuclear Regulatory Authority (ARN) and the National Social Security Administration (ANSeS). June, 2021	*Establish reciprocal collaboration between both parties, in order to incorporate into the "ANSeS goes to your Work Program (AVATT)", which aims to create a direct communication channel between both organizations.*
Agreement between Gente Sana Civil Association, Institute of Medicine and Radiomedicine and the ARN. September, 2021.	*Agreement on the bases for the adoption and development of different coordination and joint cooperation measures, in the areas that are of common interest to the parties.*
Agreement between University of Buenos Aires, Faculty of Engineering and ARN. September, 2021.	*Cooperation on areas of mutual interest.*

Table 3.8.3. – Agreements with Foreign Organizations

AGREEMENT COUNTERPARTS	AGREEMENT OBJECTIVES
Arrangement between the Nuclear Regulatory Authority (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 Nuclear Regulatory Authority, 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. REGULATORY IMPLICATIONS OF THE COVID-19 PANDEMIC

In the context of the pandemic declared by the World Health Organization associated with the transmission of the coronavirus (COVID-19), the National Government implemented immediate measures to deal with the emergency, giving rise to the publication of Decree No. 297/20 by which the "SOCIAL, PREVENTIVE AND MANDATORY ISOLATION (ASPO)" was ordered, during the period between March 20 and 31, 2020. The ASPO was extended without substantial modifications until June 2020 through Decrees No. 325/2020, 355/2020, 408/2020, 459/2020, and 493/2020.

The aforementioned decree regulated the way in which people should comply with the ASPO. The decree also detailed the people who were exempt from complying with the ordered isolation, for carrying out tasks considered essential.

Likewise, the implementation of minimum teams was required in order to ensure the operation, control and maintenance of facilities for the generation of electrical energy, of oilfields, and oil and gas refining plants, of medical facilities and of the transportation and distribution of fuels and medicines among other essential products.

Personnel not aimed to essential tasks were obliged to refrain from going to their places of work and to remain in the residence where they were located; they had to continue carrying out the tasks that were requested from their place of isolation, through the modalities established by their respective employer.

On June 7th, 2020, Decree No. 520/2020 was published, which established the conditions under which urban conglomerates could relax some of the measures implemented according to Decree No. 297/2020. The new decree established the epidemiological parameters and the availability of health infrastructure required to carry out activities considered non-essential according to protocols authorized by the provincial and national health authorities. The new isolation conditions were labeled as "PREVENTIVE AND MANDATORY SOCIAL DISTANCING (DISPO)". Both systems, ASPO and DISPO, coexisted, reversible depending on the prevailing epidemiological conditions in each place, with responsibility shared by the national, provincial and municipal authorities regarding the isolation system adopted. ASPO and DISPO conditions were successively extended and updated.

The ARN continued to work to ensure that this situation does not compromise radiological and nuclear safety, safeguards, protection and physical security in the field of nuclear activities and applications of ionizing radiation under its jurisdiction that are carried out within Argentine territory, guaranteeing, in any case, the functioning of the Institution. Consequently, the ARN has adjusted its activities by defining alternative forms of work that allow it to continue fulfilling its mission: to protect workers, the public and the environment against the harmful effects of ionizing radiation; radiological and nuclear safety in the operation of nuclear power plants, reactors and radioisotope production facilities for use in medicine and other uses; work to prevent the commission of intentional acts that may lead to severe radiological consequences or the unauthorized removal of nuclear materials or other materials or equipment subject to regulation and control under the provisions of the law, compliance with current regulations in the transport of radioactive material and in activities associated with the import and export of radioactive materials.

To sustain regulatory control, in addition to focusing its on-site inspection program on relevant facilities that carry out activities considered essential, ARN continued the regulatory control of other facilities by implementing remote document review and establishing new work modalities for the Councils of Advisors for the licensing of personnel of regulated facilities with responsibility for radiological and nuclear safety.

ARN also held as a priority function the issuance of import/export authorizations for radionuclides for the transport of radioactive materials, meeting the needs of the region's medical and industrial sectors.

The presential work of the minimum number of essential people was foreseen to guarantee the following:

- *Rutinary radiological and nuclear safety inspections continued at the CNA I, CNA II and CNE Nuclear Power Plants and at other facilities considered relevant.*
- *The staff of resident inspectors at the nuclear power plants adapted their activities to control the safe operation of the nuclear powers plants and the tasks associated with programmed outage and non-planned shutdowns. Particular technical aspects of radioactive waste management were verified on site, respecting compliance with the requirements and special measures adopted in the framework of COVID-19.*
- *Safeguards inspections were carried out at the facilities that use nuclear materials planned jointly with the Brazilian-Argentine Agency for Accounting and Control of Nuclear Materials (ABACC) and the International Atomic Energy Agency (IAEA) in compliance with the international commitments assumed by the country .*
- *Environmental control was maintained through the corresponding monitoring activities, the operation of laboratories and the corresponding evaluations.*
- *Operation of the IAEA monitoring stations for the detection of nuclear tests (CTBT) continued under the responsibility of ARN in Argentine territory.*

Through the Resolutions of the Board of Directors of the ARN No. 82/20, No. 103/20, No. 160/20, No. 219/20 and No. 313/20, the validity of the authorizations granted by the ARN (licenses of operation, registrations, non-routine practice authorizations, individual permits, specific authorizations, radioactive material transport approval certificates) was extended. Likewise, the established terms for the renewal of all authorizations granted by the ARN (operating licenses, registrations, non-routine practice authorizations, individual permits, specific authorizations, approval certificates for the transport of radioactive materials) were suspended, not being necessary to forward the required documentation unless the ARN expressly requests it.

The Board of Directors coordinated with the secretariat of the Advisory Council of the Personnel of Relevant Facilities (CALPIR) the extension of the validity of all the specific authorizations granted to the personnel of the nuclear power plants and facilities considered relevant until December 2020, making the terms for their renewal more flexible, addressing the existing difficulties during the ASPO for carrying out psychophysical and technical evaluations associated with said procedures.

As of June 2020, the treatment of individual permits for medical and industrial applications was normalized, implementing the teleworking methodology with external specialists who work on the Advisory Council for the Application of Radioisotopes and Ionizing Radiation (CAAR).

The annual retraining of the licensed personnel of Class I facilities was authorized to be carried out virtually, with a combination of presential/virtual exams depending on the facility, and with the active participation of the inspectors of each group.

All the authorizations approved in this period were digitally issued, allowing the optimization of the resources and the times associated with their management.

Regarding the recognition of valid courses for obtaining individual permits for different purposes of use of radioactive material or ionizing radiation, the introduction of virtual modality for the dictation and evaluation of the required radiological safety content was evaluated on a case-by-case basis.

In addition, to adapting its activities to alternative forms of work to avoid contagion and guarantee the health and safety of its employees, ARN confirmed that NA-SA and CNEA were adequately prepared to continue their safe operation under the restrictive conditions imposed for the ASPO.

Continuity was given to the personnel dosimetry service and to the maintenance and calibration of inspection equipment and laboratories.

In order to guarantee the necessary radiological safety and physical security conditions, the ARN carried out the corresponding actions for insurance of sources of institutions that interrupted their normal operation and could not take care of these in a timely manner.

ARN radiological and nuclear emergency intervention system was maintained to respond to possible situations that could compromise control over all radioactive or nuclear material. Also, the operation of its laboratories to support environmental and dosimetric evaluations and the operation of the nuclear test monitoring stations under its responsibility was guaranteed.

Likewise, the communications media and the radiological and nuclear emergency response system were kept active with a regimen of permanent rotating shift team. Radiological emergencies that arose were attended to.

Virtual courses were held for First Responders to radiological emergencies, which involved personnel from different response organizations (Civil Defense, Volunteer Firefighters, Medical Emergency Services, Police, etc.).Training (using e-learning tools) was given to educational institutions in the towns around nuclear power plants.

The guidelines and work modalities of ARN were adjusted to the regulations issued by the National Government through its regulations and to the recommendations of the Ministry of Health, the Ministry of Labor and the Superintendence of Occupational Risks.

Through the Resolutions of the Board of Directors of the ARN No. 190/20, it was approved the General Protocol on Prevention and Safety Measures for Health Emergency by COVID-19, that establishes indications and prevention measures to be complied with in order to protect ARN workers developing activities or essential services in person during the ASPO, as well as in response to a possible gradual return to presential work. This Protocol was prepared with the participation of the Work Conditions and Environment for Public Sector Commission (CyMAT) and an interdisciplinary group of ARN professionals, aligned with all the requirements of the Ministry of Health and other competent agencies.

Accordingly, workers that should be considered within the risk groups were identified. Likewise, the cases of contagion were counted and their follow-up was carried out.

Documents of the quality management system were adjusted/adapted, in order to consider the changes derived from the new work modalities. Internal audits of the quality management system were carried out virtually.

The computer support sector facilitated the remote connection of personnel who maintained their activity in the telework modality, and for group work meetings, videoconferences were used from different platforms (Zoom, Skype, MS Teams, Google Meet, etc.).

ARN, despite the temporary suspension of presential work, implemented different measures and actions in order to ensure the fulfillment of its functions, as well as the management of remote procedures, payment of salaries, management of resources and supply of necessary goods and services.

Communications with the regulated parties, according to the modality adopted considering a graded approach, continued via face-to-face, telephone, email and videoconferences. Also, system of formal communications was maintained. The means of communication in each case were published on the external website of the ARN.

The ARN strengthened its online communication channels during the COVID-19 pandemic. On its website, the ARN prioritized information on suitability of activities, new contact channels and resolutions of interest to regulated parties, with the creation of a new section that centralized all the news on regulatory functioning during COVID-19. Social networks also supported communication online, with an important increase in their interactions.

3.8.9. COMPLIANCE WITH THE OBLIGATIO|NS 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 by 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 may *cause* 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 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 *applicable*, 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 Licensee responsibilities, *whose most important aspects are as follows:*

- 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 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 *to* 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 Embalse Nuclear Power Plant. The *most* relevant mechanisms applied by NA-SA are the followings:

3.9.3.1. COMMUNICATION CAMPAIGNS

Mass media campaigns are annually developed, *specially focused on the local media of the influence area of the stations*. 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 in 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 in 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.

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. In addition, NA-SA communication also gathers strength through social networks such as the presence *on* Facebook, Twitter*, Instagram, LinkedIn* 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 the nuclear power plants has been carried out for years, in which more than half a million visitors have participated. During 2021, the program was adapted with a mixed virtual and face-to-face modality due to the sanitary conditions of the COVID-19 pandemic. The general public as well as different institutions like national and international agencies representatives, schools and universities continue participating in 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.

According to that, and considering the projects that are being carried out in Buenos Aires province, the local enforcement authority of the environmental impact is the Ministry of Environment of the Buenos Aires Province.

Regarding the licensing process for both the Long Term Operation of CNA I and the fourth NPP project (CNA III), NA-SA is managing the necessary actions derived from current regulations.

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.

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 *sixty* days before the date of execution. *However,* 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 by 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 life 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 nonproliferation 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 of 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 fulfils 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 complies with the provisions of the following documents:
    - Operating License.
    - Safety Analysis Report.
    - Policies and Principles Manual.
    - *Technical Specification*.
    - Operating Manual.

  - Maintenance Manual.
  - Quality Management System Manual.
  - Radiological Safety 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 recommendations 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-assessment, 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 of the plant personnel or for those members of the staff who perform safety related tasks.

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.

- *Accountability 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 principles described above are 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.

Also, this Committe revise the results of:

- Periodic evaluation of the installation performance.
- Periodic evaluation of the training personnel program.
- Evaluation of Emergency plan (including the corresponding exercises), etc.
- *Performance indicators programme.*

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 actions 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 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.

      The communication and dialogue described above are reached by on the job coaching of personnel. Coaching is considered one of the most powerful attributes for effecting change. Coaching helps influence "the people variable" in the change process towards a professional regulator-operator relationship.
- Efforts to improve the safety of the NPPs have the highest priority in both ARN and NA-SA.

As it was mentioned in previous report, NA-SA have 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. *The programme is based on work teams with participation of leaders and individuals with high credibility in the organization. The PRACS consists of selecting specific topics in the organization (fire protection, emergency preparedness, equipment reliability, human performance, risk assessment, corrective actions, indicator management, among others), through the definition of leaders responsible (coordinators), in the different NA-SA sites, who lead these issues, which must implement a specific methodology (meetings and deliverables) to carry out transversal objectives to the organization, through the fulfilment of specific actions. The status of the issues is monitored in the framework of a PRACS committee (at a plant site and headquarters level) which is led by the corresponding site manager (in the case of headquarters, the committee leader is the General Manager).* The measurement of the evolution is done through, surveys, self-assessment and performance reports.

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.

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).

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 by 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 *2019-2021* 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.

In this period, the average increase of electrical energy demand was *about 3% per year,* while the installed capacity had an *average annual* increase of approximately *4%.*

Referring to nuclear generation, CNE began its life extension on January 1st, 2016, *having completed the work in 2019.* The works that were carried out have increased the power of the plant by *8 MW* and will allow it to continue generating power for another 30 years since 2019.

Figure 3.11.1. *shows* the electrical generation from 1975 to *2021*, classified in hydroelectric, thermal, renewable and nuclear.

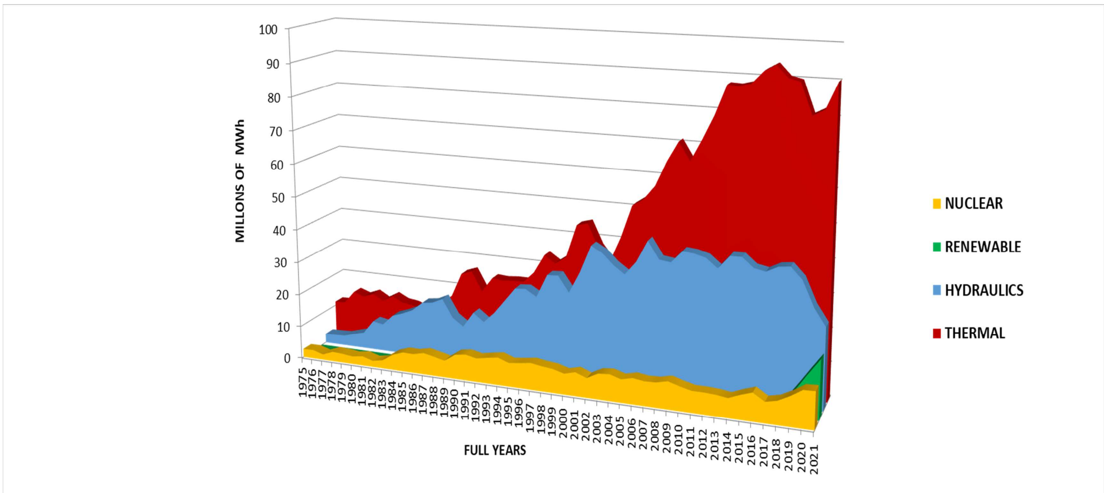


Figure 3.11.1. – Electrical generation in Argentina from 1975 to *2021*

3.11.2. ELECTRICAL GENERATION AND ECONOMICS

3.11.2.1. TOTAL ELECTRICAL GENERATION PERIOD *2019-2021*

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

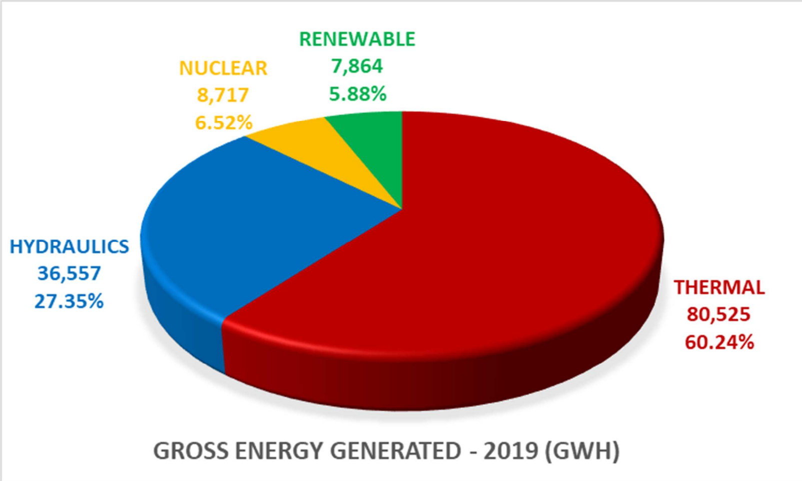


Figure 3.11.2. – Gross energy generated in *2019*

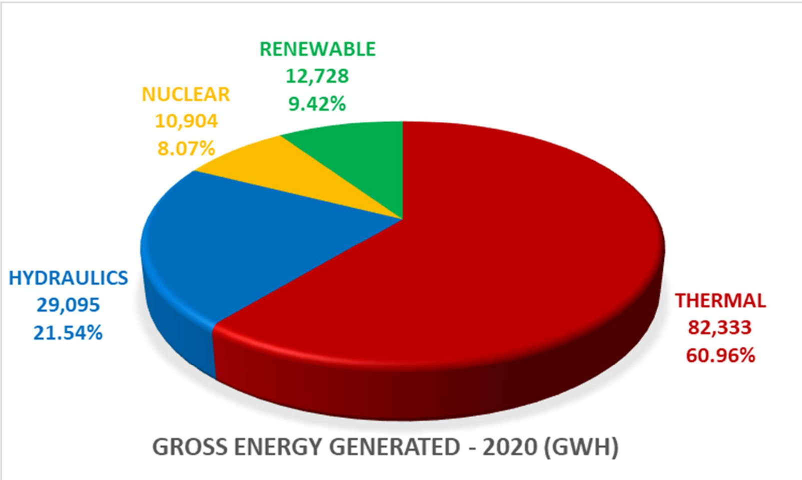


Figure 3.11.3. – Gross energy generated in *2020*

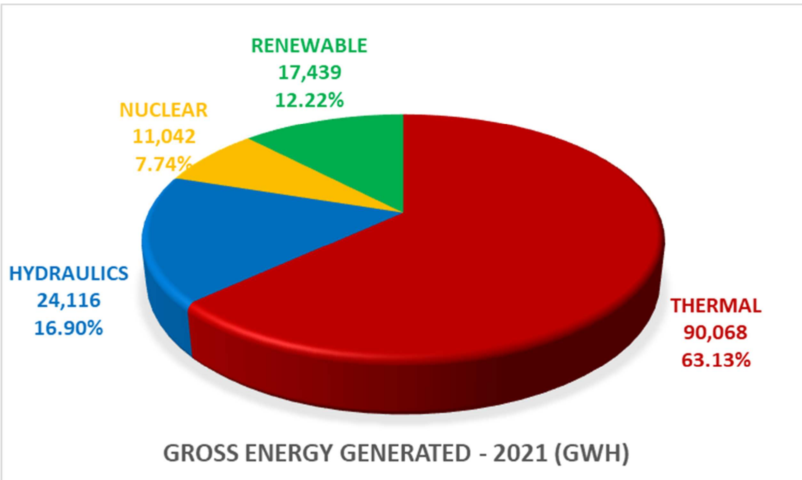


Figure 3.11.4. – Gross energy generated in *2021*

3.11.2.2. NUCLEAR GENERATION *2019, 2020, 2021*

Performance is shown in Table 3.11.1.

Table 3.11.1. – Performance

	2019	*2020*	*2021*
Gross Energy (MWh)	*8,716,966*	*10,903,734*	*11,042,083*
Load Factor (%)	*56.44*	*70.41*	*71.50*
Installed Nuclear Power (%)	*4.42*	*4.18*	*4.08*
Generated Nuclear Power (%)	*6.52*	*8.07*	*7.74*

3.11.2.3. HUMAN RESOURCES

In recent years, the personnel retirement turned out to be the main cause of loss of human resources. This is the reason why NA-SA personnel decreased in 2020 from previous year. Very few resources were lost due to the exodus of workers from NA-SA to other industries.

To fix this, NA-SA has been hiring personnel, who were trained by the experienced staff nearing retirement, since 2020, when work began on a staffing plan for the next three years (2020-2023), including over 400 new staff hires, which are being carried out as expected. As a result of this plan, the number of workers at NA-SA increased in 2021, as shown in Table 3.11.2.

Table 3.11.2. – Personnel *2019, 2020 and 2021* by work area and specific knowledge

	Year	CNA I-II	CNE + PEV	*Project IVCN*	Main Branch	TOTAL
Professionals	*2019*	*325*	*308*	*176*	*207*	*1016*
	2020	*305*	*298*	*193*	*216*	*1012*
	2021	*324*	*287*	*200*	*228*	*1039*
Technicians	*2019*	*721*	*372*	*84*	*17*	*1194*
	2020	*827*	*425*	*112*	*4*	*1368*
	2021	*981*	*530*	*107*	*17*	*1635*
Administrative personnel	*2019*	*314*	*205*	*61*	*91*	*671*
	2020	*166*	*139*	*58*	*94*	*457*
	2021	*122*	*80*	*43*	*90*	*335*
Total	*2019*	*1360*	*885*	*321*	*315*	*2881*
	2020	*1298*	*862*	*363*	*314*	*2837*
	2021	*1427*	*897*	*350*	*335*	*3009*

Several training courses have been organized through specialized centres, such as the *Dan Beninson Institute* (*National University of San Martin*) which provides training in nuclear reactors and fuel life cycle.

NA-SA signed agreements with universities such as the National University of Rosario and *Torcuato Di Tella University,* 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 engineering studies write 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 *National Council of Scientific and Technological Research* (CONICET) to sign an agreement between both institutions. Nowadays, NA-SA is analysing the formalities to elaborate the document.

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

3.11.2.4. CURRENT EXPENSES *2019, 2020, 2021*

Figure 3.11.5 shows the evolution of Operating and Maintenance costs

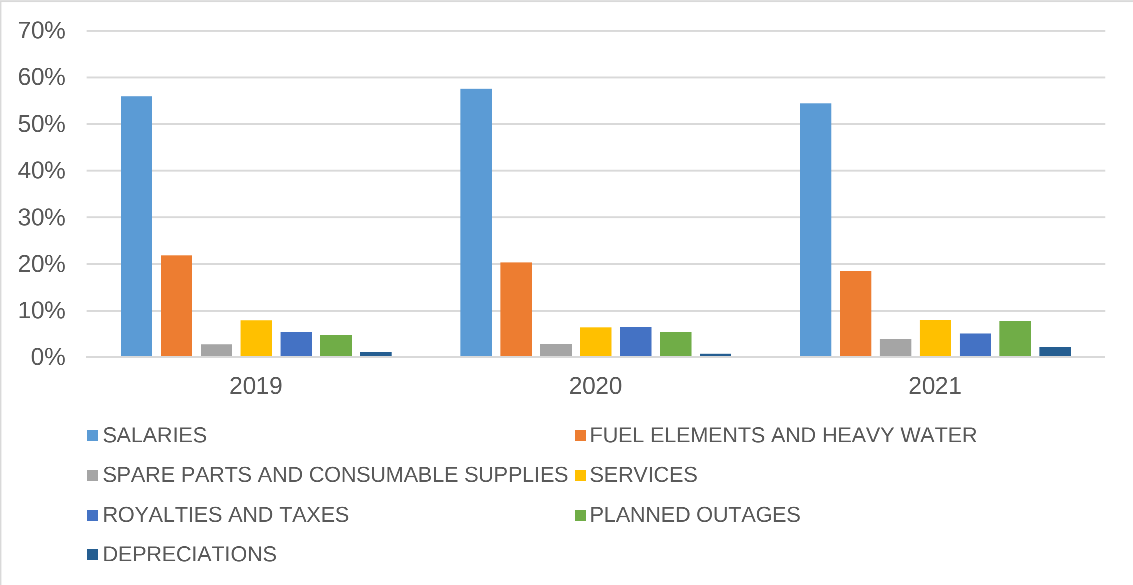


Figure 3.11.5. – Evolution of O&M costs, period *2019 – 2021*

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 by 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, *and* 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. *The reinforcement of expectations and practices of field work complements the requirements regulated by the initial and continuous training programs.*

Regarding analysis of incidents, the process acts on abnormal or unexpected events that *may* happen in the installations (operating experience, OPEX). Such events are unique opportunities to detect and *consequently* correct human errors, *by* identifying the deficiencies regarding organization, persons, 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 *of* a good root cause analysis 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 *such an* event. Finally, this analysis is evaluated to ensure that it meets the quality standards intended by the organization.

For global and systematic assessment of the installation safety, PSA *techniques are* used, *some of which consist of* the identification of human actions and the evaluation of their relative importance on the installation safety. The errors can be classified in pre-accidental (errors occurred during periodic tests or maintenance tasks) and post-accidental (errors occurred during *the conduction of* 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 *the* 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 CNE's *Full Scope Simulator*, CNA I personnel at Angra simulator in Brazil, CNA II personnel in its own simulator, as well as the human reliability analyses carried out for CNA I and CNE PSAs in shutdown states, it may be concluded that important steps have been taken to ensure that the capabilities and limitations of human performance are *properly* 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 *that it must be ensured that the operator has the necessary information at all times in order to make safety related decisions, operator action should not be required for an appropriate period after an initiating event has occurred and the operator must never be able to prevent the necessary functioning of safety systems. In addition, these regulatory standards list* the characteristics *that* the man-machine interaction *must have* in relation 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 *have* qualifications, knowledge and experience about domestic NPPs. Also the new personnel at any level are trained with the necessary knowledge before *starting to work in* the nuclear *field*.

3.12.1.1. HUMAN CORRECTIVE ACTIONS IN CNA I AND CNA II

Many human action improvements were described in previous reports. *The improvement in human behavior is monitored by the presence of leaders in the field and by the analysis of trends in internal events on CNA, with special emphasis on those internal events with causes associated with behavior. Monitoring these trends and the general actions for common findings of different internal events is the way to monitor and improve safety in work practices.*

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

- Evaluation of low *level* 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 *recurrence* 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 *level* 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.
- 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 *the* 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.

The processes of Task Observation and proactive application of error prevention techniques to identify error precursors and counteract them are introduced proactively as tools to prevent the occurrence of the event or have it of a lower level. When the event occurs, these processes are fed back from the operating experience to strengthen their proactive approach.

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 previous 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 *revisited* General *Behavior* Expectations for the *CNA* Site, developing the specific expectations of each Sub-Management and Department, *and* providing a structured framework of expected *behavior*. They are communicated through presentations by the Site Manager, refresher training courses by senior officers of the *CNA* Site, weekly reinforcement of human performance *and* reinforcement of expectations through the digital and printed billboard system available throughout the CNA *S*ite.

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 *received training* in their application:

- Pre-*work* briefing and Post-*work* debriefing.
- Procedure*s* 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 *regarding* the different techniques and methods of application in the field of action in order to minimize the presence of human error.
- *Training in Operations and Maintenance simulator of real scenarios to train human skills in order to detect error precursors and counteract them, and strengthen the correct use of the different error prevention techniques. The scenarios are developed jointly between HHFF and the trained area, and common field situations and situation in which an event occurred are used.*
- Annual Agenda of weekly reinforcement of different techniques, incorporating new concepts *related* to the techniques used, *disseminating* successful operating experience as a result of adherence to technology, as well as Plant events resulting from non-adherence. This information is weekly *disseminated* and *it* 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*s* to reverse such trend and monitoring.
- *As an immediate measure for events in which a human error was part of its causes, it is ensured at the time of dealing with it, that actions aimed at monitoring and verifying from leadership are included, and that the corrective actions undertaken are effective.*
- 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*s*.
- 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 (*such as* PRACS).

It is noted that training and retraining in human performance is defined in the initial and continuous training programs for each position. Additionally, there are 52 weekly human performance reinforcements which are disseminated by each group leader, and in which different topics are *refreshed, previously communicated expectations are reinforced, results of previous performance are disseminated, and expectations that are pursued in current performance are reiterated.*

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 analysis of internal events foresees the extension of the condition, so that, based on the analysis of an event, similar latent errors-risks can be detected and corrected before they occur.

The process and method of calculation for identifying trends, has been formalized through the development of procedures and internal instructions of the *CNA* 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 *also* 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-*work* briefings, its insertion into work packages is considered part of this program and *it* will be rigorously applied in work management process of the *CNA* 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.
- *Acknowledgement of the leaders who meet their expectations by the Site Management. .*
- General training to all management lines on issues related to team leadership.
- *Observation of observers by programme referents to ensure the effective performance of the process and the sustained introduction of improvements in the field.*
- 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).
- *Integrated system for recording and managing task observations and tracking immediate corrective actions taken on each observation.*
- Strengthen the *Plant* standards and expectations.
- To ease communication between chief-staff and job executors.

Bimonthly, *analysis* are *carried out* with each Sub-Management and half-yearly with Management to monitor compliance with the program expectations; the necessary corrective actions are formalized in the database of the site for further follow up.

A new review has been carried out to the procedure that formalizes different changes introduced to the process. Training is being performed in 2022.

Among the main changes the following can be pointed out:

- *Introduction of our observation cards to improve registration and increase the type of proactive findings that contribute to reduce the number and level of events.*
- *Description of the observer observation program.*
- *Description of the strategic observation program of the site (group of observations that, for the fourth consecutive year, are carried out in the plant and respond to a specific common theme which is transversal throughout the entire organization and defined by the top management of the site).*
- *Description of the cross-observation program (group of observations in which a leader observes a process that influences his/her area, but for which he/she is not responsible, and proposes improvements to it, registers and gives feedback to the observed group).*

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 *the* site documentation.

The management is part of the different areas allowing information to be obtained from the findings of the different observers of the site. Furthermore, answers are requested to feedback the process and commit everybody to the quality of their observation, feedback to the observed team and registration of what happened in the integrated system of task observations of the site.

In 2022, 50 new observers will be added who will contribute with their findings and coaching to improve the human performance of the Site.

3.12.1.2. 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 that arise from 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-work* briefing and *post-work* 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.
- *CNE's Human Performance Simulator*
- Self-Assessments.

3.12.1.2.1. 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-work briefing and Post-work 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 instill 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 has an Annual Agenda for weekly reinforcement of different techniques, incorporating new concepts in relation to the techniques used and expectations of management related 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).

3.12.1.2.2. 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).

3.12.1.2.3. CNE's Human Performance Simulator

In 2020, the Human Performance Simulator was built and put it in operation with the purpose of improving the processes of continuous training, qualification in human performance and expected behaviours of the staff of the Embalse Nuclear Power Plant site.

The "Error Prevention Techniques" manual, and the "Use of Personal Protection Elements" and "Radiation Protection and ALARA Recommendations" procedures are used for the creation of the scenarios, with the incorporation of intentional errors by the instructors. 605 hours of simulator use were recorded in 2020 and 2021.

3.12.1.2.4. 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 Safety.

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

In any case, it is stated that the Primary Responsible together with the Licensee, consider that the prevention of undesirable events is not achieved through a system of sanctions. Instead, it is based on the activities mentioned throughout this article, and the application of sanctions is the last resort used, depending on the severity of the infraction or error.

Nonetheless if such events occur, the NPP Primary Responsible determines the responsibility degree, if any, of persons who may have incurred in errors, and applies the corrective measures and, only when 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 NPPs global safety, identifying deficiencies in the operator actions and providing whatever is needed to analyse and perform possible corrective actions *for all plant operating states (full power and shutdown states)*.

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 *following* documents:

- 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)*. These human actions include errors in* 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 *HA* Category C *are* 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 CNA II 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*, CNA II* and CNE PSAs were based on generic data for the human error *probability*, recovery *factors* and uncertainty factors*, as well as interviews with Main Control Room personnel, Operation manuals and deterministic and thermohydraulic studies, which have* provide task execution times, performance frequency for components and equipment recovery times, *maximum time available for diagnosis and set of compelling signals of an abnormal event*. *The training of operators in the CNA II Full Range Simulator regarding the conduct of accidental sequences of certain initiating events and has also been fundamental in CNA I HRA.*

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 *by* reviewing normal operating procedures and emergency procedures*, taking* into account the PSAs results.
- *Development and implementation of the* Human Performance Programs of CNA I, CNA II and CNE.
- 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 *regard*, 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 *the* 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 by 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. 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.

In 2019, Regulatory Standard AR 10.6.1. "Management System for Safety in Nuclear Installations and Practices" was issued, so the licensees are reviewing the documentation to adjust it to the requirements of said standard.

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), Nuclear Projects Unit (UGPN) 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 evaluating the implementation of the QA Program in CNA I, CNA II and CNE, the Nuclear Projects Unit and the Maintenance Support Unit. The results of these evaluations are reported to the *President of the Board of Directors, the Executive Director, the General Manager and the Managers involved in the audit.*

Anually, the QA Management issues a report showing *the evolution of audits during the year.* These reports are sent to NA-SA's President of the Board of Directors, the Executive Director and the General 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	*42*

CNA I-II	Quality Assurance Manual for the Operation of CNA I and CNA II, Approved April 2014	*198*
CNE	Quality Assurance Manual for the Operation of CNE Approved March 2019	*234*
UGPN	Management System Manual Approved May 2016	*27*
SPC	Management System Manual Approved March 2016	*27*

The *UGPN* Management System Manual complies with Regulatory Standard, 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. *As a result of the COVID-19 pandemic, an extension to the certificate was obtained until March 2022.*

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-II;* activities related to CNAUI-LTO Project; electric power generation and Cobalt 60 production in CNE; management of nuclear projects by UGPN Unit; technical and administrative management at Headquarters. *The validity of the certificate is three years 2018-2021. However, based on the extraordinary circumstances related to COVID-19, and the provisions of the international organizations that regulate certification schemes, IRAM has decided to extend the validity of the certification until February 2022.* 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 Standards, 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 – CNA II – CNE

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 *"supplier evaluation"* is applied.

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 *issued* in 2009*, last reviewed in 2021.*

The Quality Manual is aligned with the requirements of AR *10.*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 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 imposed by Article 13 of the Convention on Nuclear Safety.

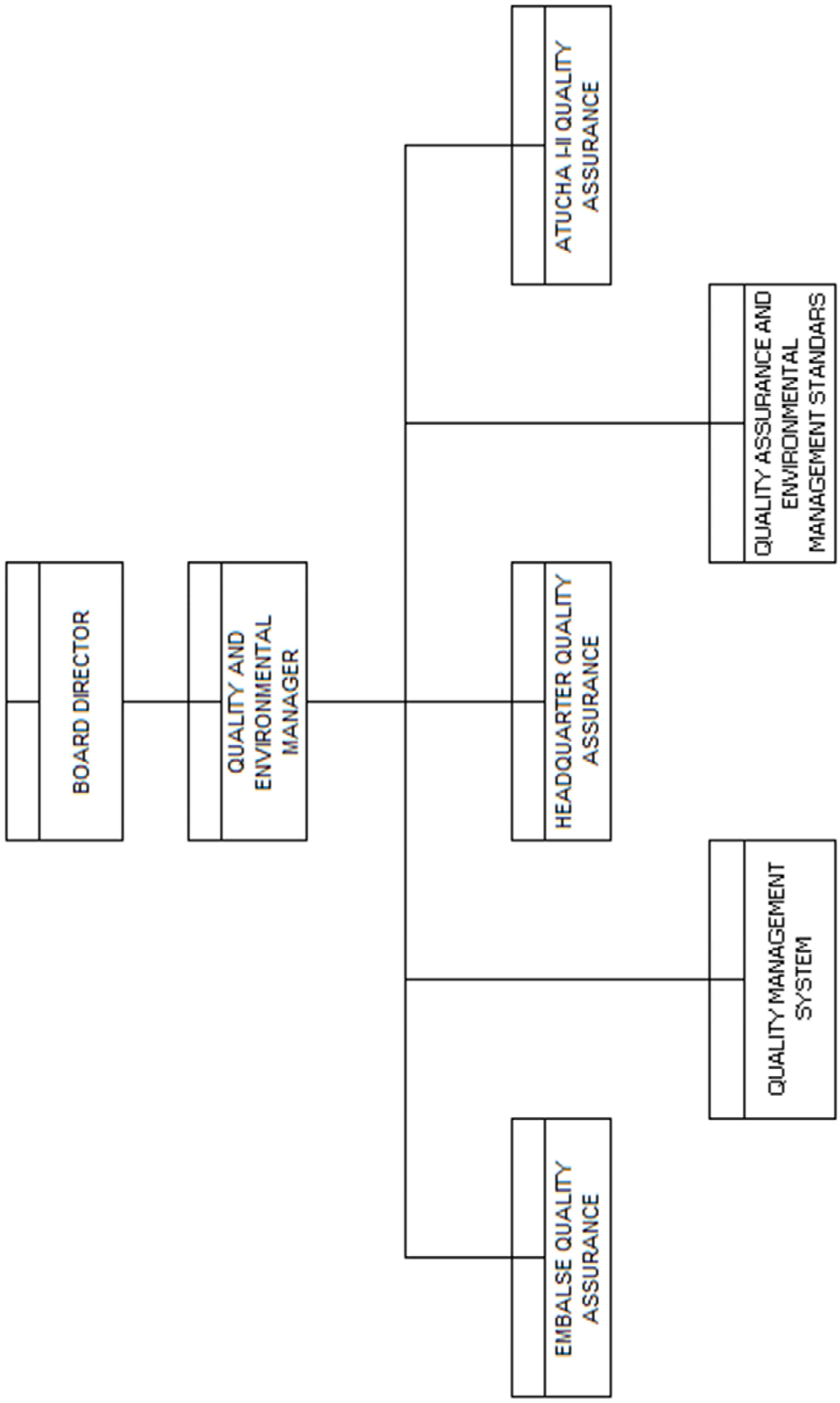


Figure 3.13.1. - QA Organizational Chart

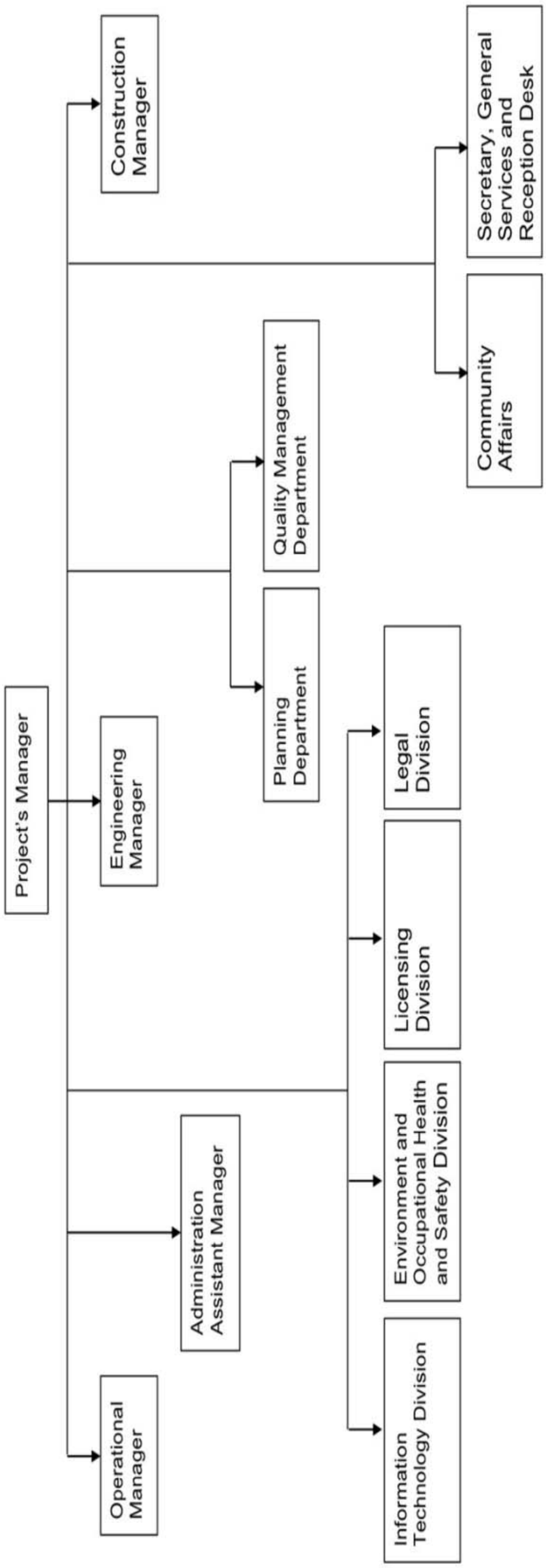


Figure 3.13.2. - Organizational Chart CAREM project

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.

Regarding the activities to be performed by the Licensee in order to verify the physical state of SSCs, AR standards as well as License's conditions include several requirements. For instance, in order to assure a reliable operation of systems and components, adequate maintenance, regular in-service inspections and periodical tests, are required.

In general, main programs used for maintaining and verification of the operability of a NPP are the following:

- *Periodic testing according to the Operational Limits and Conditions.*
- *Preventive and predictive maintenance programs.*
- *In-service inspection programs for pressure retaining components.*
- *Surveillance program of reactor pressure vessel material.*
- *Other research programs for evaluating the ageing of components and materials.*

For each of the existing NPP, the safety assessments made during the design, construction, commissioning and operation stages up to *2019* 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. SAFETY ASSESSMENT

Safety assessments cover all the plant operating states, 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 3.14.2.1. and 3.14.2.2.

3.14.2.1. 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 Assessment, for Severe Accident Analysis and for specific requirements, such as the stress test analysis after the Fukushima accident.

3.14.2.1.1. Safety Analysis Upgrading

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.

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). *A new version will be performed and submitted to Regulatory Body for License Renewal before starting Phase B of long term operation which is forseen not earlier than 2024.*

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 the life extension.

Related to CAREM 25 prototype reactor, the deterministic safety analysis was already submitted to ARN and it is currently under regulatory review.

3.14.2.1.2. 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 previous National Safety Reports, as result of the evaluation several improvements were required by the Regulatory Body.

Deterministic assessments were developed or are under development in order to support the elaboration of the Severe Accident Management Guidelines (SAMG) according to the strategies proposed in the framework of the Severe Accidents Management Program.

In the following sections the status of the activities with respect to severe accident management, are described.

3.14.2.1.3. Accident Management and Severe Accident Management Program

As was mentioned in previous National Safety 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 *uncover* initiation.
- Strategies proposal
    - 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:

3.14.2.1.3.1. 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.*2*.1.3.4.

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. *These results were taken into consideration in the update of Guidelines and were included in the SAM documentation.*

As far as Primary System Feed & Bleed is concerned, the results obtained were used to dimension the new PORVs in CNA I, to be installed during the Long Term Operation project.

Full PSA Level 2 and *3 for Atucha I NPP will be agreed with* the ARN. 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, *both in the containment and the annulus building.*

This model *was also used* to asses some preliminary SAM Strategies for Containment. In particular, *some bounding calculations to assess the need for a Filtered Containment Venting System (FCVS) for Atucha I were performed.*

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” were 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.

During the years 2019-2021 a set of analysis for CNA II were performed with MELCOR Code, to assess above mentioned issues (refer to section 3.14.2.1.3.2 for details). The open literature suggests that when it comes to severe accident analysis, if dedicated models for the plant being assessed are not available, models and conclusions for NPPs of similar design can be extrapolated.

Since MELCOR Model (which would allow Containment analysis and long term behavior during a severe accident) is not yet available for CNA I, the results obtained for CNA II were extrapolated for CNA I. These results indicate that in the case of a severe accident without containment bypass through Safety Injection Pumps suction lines, the pressurization of the Containment is very slow, reaching Design Pressure only after about 4 days, at the earliest and in the worst scenario.

Furthermore, a deep study of international requirements was performed, with regard to Generation II NPPs. In particular, WENRA Report on Safety Reference Levels for Existing Reactors states in Section F4.1 that the objective of DEC-B is "that the plant shall be able to fulfill confinement of radioactive material". The irreversible loss of the confinement function and the associated uncontrolled consequences should be avoided.

In CNA I, the main issue that threatens the confinement safety function is containment bypass through Safety Injection Pumps suction lines. Installation of a FCVS does not avoid containment bypass and would be beneficial to decrease dose to public only in the unprovable events without failure of the SIP suction lines.

Therefore, it was decided that this design flaw should be assessed and prioritized during the Long Term Operation project. This will be done by the installation of so-called "Corium-Barriers".

Cooling of the RPV external side

The RPV external side cooling *was required by the regulatory body to be considered* as a means for retaining the corium in scenarios with extensive core damage. The strategy and its effectiveness *were under analysis during the years 2016-2019.* In the past years, preliminary results obtained with RELAP5 / SCDAP were performed. Also, more complex analysis with ANSYS / CFD code has been performed for CNA II NPP.

Regardless the results of these analyses are not yet conclusive*, this countermeasure has been ruled out, because it was demonstrated that it's successfulness in the full set of Severe Accident could not be ensured.*

Corium-Barrier to improve Containment long term integrity

During the last years, NA-SA started to assess possible stabilization of molten material inside sump, to avoid containment breach due to Containment bypass through SIP suction lines, or at least delay to it, so as to decrease consequences in public as far as reasonable achievable.

In CNA I, the sump design, volume, relative location of RPV and SIP suction lines, is such that in the event of a Severe Core Damage accident, if total meltdown of the core plus main internals is considered, the volume of corium can be contained inside a set of walls that would avoid direct contact with SIP suction lines.

The material specification of such walls, as well as the determination of expected integrity is under development by NA-SA and CONICET (specialists in ceramics materials have been contacted).

The underlying concept of Corium-Barriers is that they are not required to remain in place in the very long term, but only as long as the containment would eventually fail by other means (e.g. overpressure due to slow pressurization). They are intended to delay containment bypass, which would eventually occur in the present design, so they should not be compared to Core Catchers in modern PWR designs.

As the concept engineering of this strategy involves highly complex studies, it was proposed to incorporate it into the Long Term Operation project.

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 3s, 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.
- *Instruction T-18 ("River Drought"): the instruction was updated in relation to the level of the Parana River*.

Instrumentation and Control

The installation of an extra level measurement in the reactor has already been implemented.

Besides, level *and new temperature measurements were* 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 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 Long Term Operation project.

3.14.2.1.3.2. 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 3.14.2.1.3.4.

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 CNA 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 *and injection to primary system (via Control Volume System)* as severe accident management strategies.

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.

During year 2021 and 2022, Revision 1 of PSA L2 is being performed. As a result of this revision, new accident progression analysis with MELCOR Model was requested. These analyses were mostly performed with the same MELCOR Model developed and used during the licensing process of CNA II.

Nevertheless, new scenarios (Steam Generator tube rupture and Moderator Heat Exchanger tube rupture as initiating events), were requested. These required input model modifications are currently under development.

Filtered Containment Venting system

During years 2020-2021, CNA II detailed MELCOR Model was used to evaluate the maximum pressure that can be reached in the Containment, in the case of a severe accident without containment bypass through Safety Injection Pumps suction lines. For this analysis, taking into consideration results obtained from Full Scope PSA L2 of CNA II reactor, four accident scenarios were analyzed, namely: Moderator LOCA without SIP, SBO, SBO in conjunction with Moderator LOCA and TLFW. For each of these scenarios, the conjunctions of additional failures leading to maximum Containment Pressure were selected.

Available MELCOR analyses were re-calculated imposing a no-containment bypass through SIP lines boundary condition, letting the Containment to go under slow pressurization. Obtained results indicate that the pressurization of the Containment is very slow, reaching Design Pressure only after about 4 days. Results were complemented by scenarios of water addition to Corium in sump, to assess the influence of steam generation in the obtained results.

NPPs in Argentina are required to fulfill AR 3.1.3. Standard, that requires that all groups of Severe Accident Scenarios remain below a curve of iso-risk (dose to the public vs. probability combination should remain under a specified value). For CNA II this analysis has been performed fully for Licensing and it shows that main three groups mostly contributing to risk are scenarios in which containment is bypassed from the initiating event (e.g., SG Safety Valve fails open, Ventilation System failure, etc.). For these type of scenarios, FCVS would, if at all, contribute only marginally to a dose decrease to the public.

Furthermore, as already mentioned, a deep study of international requirements was performed, with regard to Generation II NPPs. In particular, WENRA Report on Safety Reference Levels for Existing Reactors states in Section F4.1 that the objective of DEC-B is "that the plant shall be able to fulfill confinement of radioactive material". The irreversible loss of the confinement function and the associated uncontrolled consequences should be avoided.

In CNA II, as in CNA I, the main issue that threatens the confinement safety function is containment bypass through Safety Injection Pumps suction lines. Installation of a FCVS does not avoid containment bypass and would be beneficial to decrease dose to public only in the unprovable events without failure of the SIP suction lines.

Therefore, given the timing after the initiating event at which the FCVS would be used, the groups of scenarios for which it would be useful (all of very low probability and far from the AR 3.1.3. limiting curve, as shown by PSA L2), the approach of WENRA, and the fact that the most stringent design flaw that should be assessed is Containment Bypass through SIP suction lines, it was decided that FCVS will not be installed in CNA II.

Confinement of radioactive material will be improved by the installation of so-called "Corium-Barriers".

Cooling the RPV external side

The RPV external cooling *was requested by the regulatory body to be* considered as a means for retaining the corium inside the vessel in scenarios with extensive core damage. The strategy and its effectiveness *were* under analysis. Preliminary results obtained with RELAP5 / SCDAP were performed, *which in principle showed that the countermeasure could be successful*. These simple calculations were followed by more complex analysis with ANSYS / CFD code, performed for CNA II NPP.

After a first set of simplified calculations, very detailed analysis with ANSYS / CFD code *were* performed for SBO scenario. These analyses were intended to assess if a very Best Estimate calculation can lead to successfulness of the strategy. The calculation *included* 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.

The results showed that after a few hours, integrity of the RPV could not be assured, since creep failure at high temperature was a threat. In practice, this countermeasure has been ruled out.

Corium-Barrier to improve Containment long term integrity

During the last years, NA-SA started to assess possible stabilization of molten material inside sump, to avoid containment breach due to Containment bypass through SIP suction lines for CNA I. This concept was then transferred to CNA II.

The design of both NPPs, though similar, is not exactly the same. In particular, CNA II sump design, volume, relative location of RPV and SIP suction lines, is different to that of CNA I. Therefore, studies are being performed to evaluate if the installation of Corium-Barriers is possible in CNA II, to avoid or at least delay containment bypass.

This is a task under development in Safety Analysis and Engineering groups with support of CONICET (specialists in ceramics materials and 3D calculations).

It must be pointed out that the underlying concept of Corium-Barriers is that they are not required to remain in place in the very long term, but only as long as the containment would eventually fail by other means (e.g. overpressure due to slow pressurization). They are intended to delay containment bypass, which would eventually occur in the present design, so they should not be compared to Core Catchers in modern PWR designs.

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.

New level and temperature measurements were 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.

Recently a full assessment of installed instrumentation and its ability to function during different phases of a severe accident has been completed and delivered to the ARN, as requested. This is the first step towards identifying the necessary modifications to the plant, so as to count with a minimum set of PAM I&C during an accident with core melt.

Alternative power sources

The analysis of essential consumption required to face with severe accident situations caused by a SBO was developed. The following activities *have been* carried out:

- Study of the connections between mobile diesel generator and electric bars.
- Definition of the power and location of the mobile diesel generator.
- Technical specifications of the mobile diesel generator.
- Detailed engineering of this improvement.

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 *and internal instructions* were drawn up within the framework of severe accident management (See section 3.14.2.1.3.4.):

- 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).
- *ORE-015: Water supply to the tanks of the GHC system - Demineralized water from the SGA fire system- Fire network.*
- *ORE-007: Power supply SGA system from the fire network of the UG-PN workshop.*

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 *the corresponding SAMG has been issued.*

Review and improvement of accident management procedures

The review of the following procedure has been made:

- *Instruction 3.03.03: New low river level. This will allow systematic manoeuvres to carry a plant outage. Adequacy of criteria in relation to the level of the Paraná River*

3.14.2.1.3.3. CNE

ECCS reliability

Improvements in the ECCS´s reliability were implemented in order 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. *During the 2021 planned outage was implemented the ECCS heat exchanger bypass.*

Implementation of emergency filtered containment venting system (EFCVS)

During CNE Life Extension, a seismically qualified EFCVS was installed to the containment building to relieve the high pressure of the building in a filtered and controlled way, avoiding breaks in the structure and the release of radionuclides into the environment in case of severe accident.

Implementation of calandria vault make-up line (CVML)

During CNE Life Extension, a seismically qualified CVML was installed to replenish inventory to the calandria vault from outside the Reactor Building (exteriors) even in the absence of electrical power and instrument air, using fire trucks, in order to avoid CCI in case of severe accident.

Implementation of a back-up gas system for the Dousing System valves of the R/B

During CNE 2021 planned outage, a seismically qualified manual back-up N2 system was installed to supply gas to the Dousing System pneumatic valves actuators from outside the reactor building

(Service Building) even in the absence of electrical power and instrument air, using N2 cylinders and PRV's, to allow the actuation of individual dousing valves on demand in case of severe accident.

Instrumentation and Control

According to what is established by the Joint Projects that CNE bought from the Candu Owners Group (COG JP-4056 and JP-4426 Post Fukushima), during a severe accident, in general, it is used the relevant instrumentation that already exists in the plant (with its respective updates, such as the environmental and seismic qualification), and only for those cases where the instrumentation did not exist in the plant, is it planned to install specific instrumentation. During the CNE 2022 planned outage, it is scheduled to complete the installation of the reactor building wide-range dose rate instrumentation, and after said outage, in the short term, the installation of the containment H2 monitoring instrumentation.

On the other hand, during the CNE Life Extension, for the case in which MCR becomes uninhabitable, the necessary instrumentation for severe accidents management (such as calandria vault level, moderator level, etc.) was duplicated in SCA.

3.14.2.1.3.4. Development of Instructions and Guidelines

CNA I-II

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. *5*): Evaluation of plant status.
- Guideline 1-GAS-SC-01 (Rev. *4)*: 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. *4*): Power failure.
- Guideline 1-GAS-SC-04-01 (Rev. *3*): Interconnection from Unit II to Unit I.
- Guideline 1-GAS-SC-04-02 (Rev. 2): Mobile Diesel Generator.
- Guideline 1-GAS-CE-05 (Rev. *6*): Feed and Bleed of Steam Generators.
- Guideline 1-GAS-SC-05-01 (Rev.*6*): Water Injection to the Steam Generators - High Pressure Way.
- Guideline 1-GAS-SC-05-02 (Rev. *6*): Water Injection to the Steam Generators - Low Pressure Way.
- Guideline 1-GAS-SC-05-03 (Rev.*2):* Water Injection to the Steam Generators by Pressurizing the Feed Water Tank.
- *Guideline 1-GAS-CE-06 (Rev. 1): Depressurization of Primary.*
- 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. *3*): Control of Containment Conditions.
- Guideline 1-GAS-SC-10-1 (Rev. *3*): Containment Relief (TL7).
- Guideline 1-GAS-SC-10-2 (Rev. *3*): Containment Relief (TL8).
- Guideline 1-GAS-CE 12 (Rev. *8)*: 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. *6*): Evaluation of plant status.
- Guideline 2-GAS-SC-01 (Rev. *4*): 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. *6)*: Power failure.
- Guideline 2-GAS-SC-04-2 (Rev. 1): Plant Refrigeration with One Electric Train Active.
- Guideline 2-GAS-SC-04-4 (Rev. *3*): Electrical Interconnection from Unit I to Unit II.
- Guideline 2-GAS-SC-04-8 (Rev. *2*): Feeding from Emergency DG from Unit I to Unit II.
- *Guideline 2-GAS-SC-04-9 (Rev. 0): Feeding of a CNA II Emergency Busbar with Mobile Diesel Generator.*
- Guideline 2-GAS-CE-05 (Rev. *5*): Feed and Bleed of Steam Generators.
- Guideline 2-GAS-SC-05-1 (Rev.*3*): Water Injection to Steam Generators (LAB/LAH).
- Guideline 2-GAS-CE-06 (Rev. 1): Depressurization of Primary.
- Guideline 2-GAS-CE-07 (Rev. *3*): 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. *3*): Water Injection to the Primary.
- Guideline 2-GAS-SC-09-1 (Rev.*4*): Water Injection to Primary (KBA).
- Guideline 2-GAS-CE-11 (Rev. *8*): 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. *3*): Containment Conditions Control.
- *Guideline 2-GAS-SC-14-1 (Rev.3): Cooling of the Containment from the inside by Air Recirculation.*
- *Guideline 2-GAS-SC-14-2 (Rev.0): UJA Containment Relief to Chimney.*

(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 (ORE):

- ORE-006: Operation of mobile diesel generator BY06D001.
- *ORE-007: Power supply SGA system from the fire network of the UG-PN workshop.*
- ORE-008: Fuel supply to emergency diesel generators CNA I-II.
- ORE-015: Water supply to the tanks of the GHC system - Demineralized water from the SGA fire system- Fire network.
- ORE-016: Fuel oil from tank EGB01BB001 to weekly tanks of emergency diesel generators.
- *ORE-025: Operation of mobile diesel generator CNA II - XKA60.*
- *ORE-026: Fuel supply to diesel mobile.*

It is noted that according to the different modifications to the facility to be carried out, the following *instruction* will be generated:

- Connecting the temporary cooling system of diesel generators.

CNE

The Severe Accident Management Guidelines (SAMG) for CNE -based on *COG-SAMG* progression analysis performed with MAAP4-CANDU code - *with a similar approach to that developed for PWRs by the Westinghouse Owners Group (WOG) - was finished in 2017, and had several revisions since then to include all plant changes that affects or improve the management of this type of accidents.* 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 program contains the following set of documents:

- *General Manual of the Severe Accident Management Program (GM-SAMP): This document contains the structure of the CNE SAMP. More specifically, it explains how the progression of damage to the core occurs in a CANDU reactor, what are the objectives and scope of the SAMP, the necessary conditions to enter and exit a severe accident, how the TSG is made up within the organization of emergency response (ORE), the missions and functions of the personnel involved and the training they require.*
- *Diagnostic Flow Chart (DFC): is intended primarily for diagnosis of plant status, and for early indication of potential challenges to the fission product boundary.*
- *Severe Challenge Status Tree (SCST): is intended to identify immediate and severe challenges to the containment boundary. This flowchart is accessed from the top of the DFC, and has priority over it.*
- *Severe Accident Guidelines (SAGs): If a plant parameter monitored using the DFC goes beyond its setpoint, the TSG utilizes a Severe Accident Guideline (SAG) specific to that parameter to identify strategies and actions to bring that parameter within an acceptable range. The TSG evaluates the positive and negative consequences of various strategies, recommends the preferred strategy for implementation by control room staff, and monitors effectiveness of the implementation based on information available to the TSG, particularly feedback from control*

room staff. It is possible that multiple SAGs will be in progress at the same time if resources allow, but the TSG must keep in mind the established SAG prioritization established in the DFC.

These guides are the following:

- SAG-1: Inject into the Heat Transport System.
- SAG-2: Control Moderator Conditions.
- SAG-3: Control Calandria Vault Conditions.
- SAG-4: Reduce Fission Product Releases.
- SAG-5: Reduce Containment Hydrogen.
- SAG-6: Control Containment Conditions.
- SAG-7: Inject into Containment.

- *Severe Challenge Guidelines (SCGs): If at any time a plant parameter monitored using the SCST goes beyond its setpoint, immediate action is required to address the situation. DFC monitoring and any inprogress SAGs can be suspended. Due to the urgent nature of the challenge, there is no requirement for the TSG to weigh the positive and negative impacts of potential implementation strategies. The TSG utilizes a Severe Challenge Guideline (SCG) specific to that parameter to identify and recommend the best available strategy for immediate implementation by control room staff. These guides are the following:*
  - *SCG-1: Mitigate fission product release.*
  - *SCG-2: Reduce containment pressure.*
  - *SCG-3: Control containment atmosphere flammability.*
  - *SCG-4: Control containment vacuum.*
- *Severe Accident Control Room Guidelines (SACRGs): Upon SAMG entry, control room staff commence following instructions provided in Severe Accident Control Room Guideline - Initial Response (SACRG-1). These instructions are an extension of the EOPs, reinforcing abbreviated EOP symptom monitoring and treatment (e.g., ensure reactor shutdown, ensure containment button-up, recover subcooling margin or moderator heat sink) until the TSG is operational. At this point, the control room staff discontinues use of SACRG-1, and commences following instructions provided in SACRG-2. The primary objectives of SACRG-2 are ensure control room staff update the SMC/TSG on the status of plant equipment and conditions, facilitate communications between control room and TSG staff in order to enhance control room staff acceptance of TSG recommendations and enable control room staff to implement strategies recommended by the TSG. These guides are the following:*
  - *SACRG-1: Initial Response.*
  - *SACRG-2: Technical Support Group Functional.*
- *Severe Accident Exit Guidelines (SAEGs): Prior to SAG/SCG exit, the TSG identifies any long-term concerns associated with use of any implemented strategies. Following SAG/SCG exit, the TSG continues DFC/SCST monitoring, and initiates monitoring of any long-term concerns using the Severe Accident Exit Guideline #1 (SAEG-1). SAEG-1 also enables a review of recovery actions when no SAG/SCG strategy is available due to equipment failure. This situation may occur early in the accident when much equipment may be unavailable. The DFC diagnosis loop mandates this review by the TSG. The final step in the DFC diagnosis loop provides criteria to determine that sufficient DFC parameters are within their acceptable ranges, stable or trending in the safe direction, and that the plant is judged to be in a controlled, stable state. If these criteria are met, the DFC instructs the TSG to initiate the process of terminating SAMG. The TSG conducts this process using the Severe Accident Exit Guideline #2 (SAEG-2). These guides are the following:*
  - *SAEG-1: Long Term Monitoring.*
  - *SAEG-2: SAMG Termination.*
- *Calculation Aids (CAs): They are analytical tools that allow the evaluation and estimation of some plant conditions or parameters when direct instrumentation is not available or adequate, and they are called from the GAS's/GDC's when the situation requires it. These aids are the following:*

- *CA-1: Individual dose to a member of the public from a containment vent.*
- *CA-2: Rate of water addition for decay heat removal by vaporization.*
- *CA-3: Rate of water addition to maintain or increase moderator level.*
- *CA-4: Hydrogen flammability in containment.*
- *CA-5: Containment water level.*
- *CA-6: Magnitude of core damage from measured dose rates.*

- *Operating Instructions (OIs): Each one of the mitigation strategies contained within the SAGs/SCGs is associated with a procedure called Operating Instruction (OI) that provides the operation shift with the specific instructions necessary to implement the strategies proposed by the TSG that have been approved by the Emergency Command. They contain in detail all the actions that must be executed in the MCR and SCA.*
- *Field Operations (FOs): when necessary and as a complement, the OIs call another group of procedures named Field Operations (FOs), which provide the field personnel of the operation shift (assistants, operators and specialists), specific instructions to execute in different buildings of the plant, that are necessary to complete the execution of a given strategy.*

Apart from the documents mentioned so far, new documents are currently being prepared that will be used in the management of severe accident in the spent fuel bays. At the moment, this type of accident is managed through EOPs.

3.14.2.1.4. CAREM Deterministic Safety Analysis

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 Analysis (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.

Moreover, deterministic safety analysis was carried out to give support to the improvement of systems important to safety, including containment ones, in order to consolidate basic engineering stage.

3.14.2.2. PROBABILISTIC SAFETY ASSESSMENT

The upgraded activities covering period between 2013 and *2022* in the applications of PSA are the following:

3.14.2.2.1. CNA I Probabilistic Safety Assessment applications

As it was mentioned in previous National Safety Report, 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 1.3×10^{-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.

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.

Revision 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 has been approved by the Regulatory Body.*

Revision 1 of the Probabilistic Safety Assessment Level 1 *was developed* for other operational states *and submitted to the Regulatory Body*. The purpose of this revision is to:

- Calculate the Core Damage Frequency 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 (SSC).
- 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.

3.14.2.2.2. 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.

During this period (2013-*2022*) the following PSA L1 tasks were *performed*:

- 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*, prioritization of preventive maintenance and evaluation of changes in the test intervals* using probabilistic techniques*.*
- *Updating the system analysis considering the actual configuration (changes from the original design).*
- *Incorporation of the detailed fault trees for the main reactor protection signals.*
- *Improvement in the modeling of common cause failures in systems with high degree of redundancy.*
- *Assessment of plant modification and accident management guides arisen from Post-Fukushima requirements.*
- *Post-accidental human actions calculation considering the final version of the Operating Manual Procedures and engineering modifications.*
- *Revision 2 of the PSA L1 in order to include all modifications and updates of PSA L1 model that were performed since the issuing of Revision 1.*
- *Internal Fire PSA.*
- *Preparation of the PSA L1 model for its implementation in a risk monitor. This real-time analysis tool will allow to determine the instantaneous risk based on actual plant configuration.*

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 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 review of PSA L2 and L3 was performed by the Regulatory Body with the assistance of Sandia N.L. and GRS during 2015.

Currently, the Revision 2 of PSA L2 and LERF calculation are under development.

3.14.2.2.3. CNE Probabilistic Safety Assessment applications

In 2004 the Licensee completed a first PSA L1 version considering full power internal events. This version 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 some proposed modification intended to quantify the impact on the risk and to obtain approval from the Nuclear Regulatory Authority to implement the change.

In another example of application, PSA study was used to assess the effect on risk due to the extension of testing period from 12 to 18 months. As part of the decision making process, the ARN required to evaluate the impact on the risk due to such increase in the time between the application of preventive maintenance, periodical tests and in service inspections.

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 and considering the real time that the components of the system were out of service for maintenance. This result is compared with an established target value for every system.

CNE finished first operation cycle at the end of 2015 and started refurbishment outage. The Licensee developed a new PSA L1 revision to update the previous one considering scope of full power internal events and including the safety system design modifications implemented during the refurbishment. This new PSA L1 version was finished in 2015.

In addition to these safety system modifications, the Licensee considered new deterministic results corresponding to transients and postulated accidents that allow the review of some accident sequences. The results 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 has 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 3.46×10^{-5} per 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 in each case. Each damage state contributor sequences coming from different internal initiating events are grouped. The CNE PSA L2 was finished in 2015.

The plant seismic safety assessment was 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.

Modifications in the plant configuration due to the PLEX were taken into account in the PSA model, for example: new headings in the event trees, new equipment, support systems and update of availability parameters.

New postulated Initiating Events were incorporated such as: General In-Containment LOCAs, Single-Channel LOCAs, interface LOCAs, In-Containment Primary Coolant Leaks, Interface Primary Coolant Leaks, Interface Moderator Discharges, General Plant Transients, Failures in SGs and in Condensate, Feed-Water and related Systems, Failures in Main Steam, Condenser Cooling and related Systems, Failures in Moderator, Shield Cooling and related Systems, Failures in Electric Supply Systems, Failures in Non-Electric-Supply Support Systems and Failures in Fuelling Machines while coupled to the Reactor.

Some human actions were added and others were modified.

Some improvements were included in the system modelling such as: ECCS, new EPSS and class 3 DGs.

3.14.2.2.4. 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.

3.14.2.3. 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 decided 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 was engaged in a plant life extension programme, PSR was 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.

Currently, ARN and NA-SA are working in the so called “PSR Basis Document” where the agreement on the general scope and requirements for the CNA II’s PSR, and its expected outcome, are documented. The PSR for CNA II will be submitted early in 2024.

3.14.3. VERIFICATION OF SAFETY

3.14.3.1. AGEING MANAGEMENT

In Argentina, management of ageing is a regulatory requirement as stated in License’s conditions.

The development of a comprehensive Ageing Management Program is one of the mandatory programs needed for granting a License and for remain valid the licensing basis of each NPP. In addition, the program has to be duly implemented in order to ensure that the degradation mechanisms *and the ageing effects do not (and will not) affect the capability of the SSCs to carry on their planned safety function, throughout the whole life cycle, taking into account the changes that occur in these SSCs due to the time and usage.*

NA-SA follows the IAEA Safety Standards and Guidelines (SSG-48 and SRS No. 82) to develop the Ageing Management Programs and perform other ageing activities in its plants. The ARN's expectations for an effective implementation of the ageing program and activities are aligned with these standards, as well.

In CNA I and CNE, due to the decision to continue in operation beyond the original design life, comprehensive condition assessment of SSCs important to safety were performed. Through these assessments the plant has examined the following aspects for each SSC:

- *Existing or anticipated ageing processes;*
- *Implications of changes to design requirements and standards, for example in materials;*
- *Plant programmes supporting ongoing confidence in the condition of the SSC;*
- *Significant findings from tests of the functional capability of the SSC;*
- *Results of inspections and/or walkdowns of the SSC;*
- *Evaluation of the operating history of the SSC;*
- *Evaluation of the external operating experience, etc.*

These conditions assessment reports resulted in several recommendations which identified the necesity of replacement, refurbishment and perform a prognosis analysis of specific components in order to mantain the plant safety desing margins.

Some activities performed in the reported period related with ageing management of SSCs are the following:

CNE

- *Ageing related degradation mechanism reports were issued for I&C components, mechanical components and civil structures respectively.*
- *A procedure to review the existing plant programs against the nice attributes was issued.*
- *Procedures for scope setting of structures and components subjet to Aging Management were issued.*
- *Improvements were implemented in plant programs such as: maintenance, ISI, based on the recomendations set on the assessment reports.*
- *The management of recomendations, raised from the assessment reports, is in progress.*
- *Since 2021, quartely informs have been issued to report the ARN the ageing improvements on each period.*
- *About cable ageing treatement. Different samples of cables were installed, inside the reactor building. Periodically, these samples will be taking out to perform ageing testing activities.*
- *Procedure for developing specific AMPs for CNE was written and it is under revision.*
- *The procedure about Cable AMP was written, and it is under revision.*

CNA I - 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.
- *Condition assessments for CNA I's SSCs were completed.*

- 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*, chemistry, etc. were* reviewed against the nine attributes of IAEA SSG-48.
- *AMPs development is ongoing.*
- Regarding management of obsolescence, an *proactive obsolescence manual* was developed based on IGALL *TOP and industry references*.

3.14.3.2. SURVEILLANCE ACTIVITIES: 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 Long Term Operation.

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 campaing was in 2021, and no relevant indications were reported. 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.*

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 defined the activities that had to 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 were:

- Transients analysis update to support the selection of 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, the CIEMAT (Center for Energy, Environmental and Technological Research) carried 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.

3.14.3.3. TESTING OF SAFETY RELATED COMPONENTS - CNE

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 were: 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 detail, 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 *2800* m^2 to 3195 m^2.
- Feedwater temperature increase from 167ºC 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.

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 by 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.

The Regulatory Body, for relevant radionuclides, set authorized values to the environmental releases of the NPPs. They were calculated considering the dose, due to all exposure pathways, for an 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 gaseous radioactive releases to the environment due to CNA I operation in the 2019-2021 period can be observed in Table 3.15.1., discriminating those corresponding to I-131, tritium, aerosols, noble gases and C-14.

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)	C-14 (TBq)
2019	8.4×10^{-6}	6.8×10^{2}	1.1×10^{-5}	1.7×10^{1}	4.3×10^{-1}
2020	1.9×10^{-4}	5.4×10^{2}	5.9×10^{-5}	3.2×10^{1}	6.9×10^{-1}
2021	2.4×10^{-5}	6.1×10^{2}	1.3×10^{-5}	3.5×10^{1}	4.0×10^{-1}

The liquid radioactive discharges to the environment by CNA I, for the same period, are presented in Table 3.15.2., discriminating between liquid discharges of Tritium and other radionuclides.

Table 3.15.2. - Activity released from CNA I to the environment as liquid discharges

YEAR	TRITIUM (TBq)	OTHER RADIONUCLIDES (TBq)
2019	2.3×10^{3}	1.7×10^{-1}
2020	1.7×10^{3}	2.2×10^{-1}
2021	6.1×10^{2}	3.4×10^{-1}

Of the total annual average discharges from CNA I to the environment, *a little bit more than 98%,* correspond to tritium.

3.15.2.2. PUBLIC EXPOSURE

It is performed a comparison of activity concentration results from environmental samples against different derived guideline or reference levels. No guideline levels were exceeded in the period from *2019 to 2022*. 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 detected in sediment and soil samples alternatively in upwind/upstream and downwind/downstream sampling points. All these results, 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)
2019	*7.39×10^{-2}*	*9.55×10^{-2}*	*1.69×10^{-1}*
2020	*6.27×10^{-2}*	*9.23×10^{-2}*	*1.55×10^{-1}*
2021	*6.64×10^{-2}*	*1.25×10^{-1}*	*1.91×10^{-1}*

3.15.3. CNA II

3.15.3.1. RADIOACTIVE RELEASES INTO THE ENVIRONMENT

The gaseous radioactive releases to the environment due to CNA II operation in the 2019-2021 period can be observed in Table 3.15.4., discriminating those corresponding to I-131, tritium, aerosols, noble gases and C-14.

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)
2019	*6.1×10^{-3}*	*9.4×10^{2}*	*1.2×10^{-4}*	*1.8×10^{2}*	*2.2×10^{-1}*
2020	*4.7×10^{-3}*	*1.0×10^{3}*	*9.4×10^{-5}*	*1.3×10^{2}*	*6.7×10^{-2}*
2021	*2.4×10^{-5}*	*6.1×10^{2}*	*2.3×10^{-5}*	*5.3×10^{1}*	*1.3×10^{-1}*

The liquid radioactive discharges to the environment by CNA II, for the same period, are presented in Table 3.15.5., discriminating between liquid discharges of Tritium and other radionuclides.

Table 3.15.5. - Activity released from CNA II to the environment as liquid discharges

YEAR	TRITIUM (TBq)	OTHER RADIONUCLIDES (TBq)
2019	*4.0×10^{2}*	*1.5×10^{-1}*
2020	*8.5×10^{2}*	*1.0×10^{-1}*
2021	*6.1×10^{2}*	*1.1×10^{-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)
2019	*7.32 x 10^{-2}*	*7.45 x 10^{-2}*	*1.48 x 10^{-1}*
2020	*7.49 x 10^{-2}*	*6.32 x 10^{-2}*	*1.38 x 10^{-1}*
2021	*9.45 x 10^{-2}*	*4.51 x 10^{-2}*	*1.40 x 10^{-1}*

3.15.4. CNE

3.15.4.1. RADIOACTIVE RELEASES INTO THE ENVIRONMENT

The gaseous radioactive releases by CNE to the environment, for the 2019-2021 period can be seen in Table 3.15.7. discriminating those corresponding to I-131, tritium, aerosols, noble gases and C-14.

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)
2019	*2.69 x 10^{-8}*	*2.48 x 10^{2}*	*1.00 x 10^{-11}*	*1.36 x 10^{2}*	*3.05 x 10^{-4}*
2020	*5.44 x 10^{-9}*	*1.21 x 10^{2}*	*1.43 x 10^{-5}*	*7.13 x 10^{1}*	*6.43 x 10^{-1}*
2021	*3.73 x 10^{-9}*	*1.35 x 10^{2}*	*6.93 x 10^{-12}*	*1.56 x 10^{1}*	*2.46 x 10^{0}*

The liquid radioactive discharges to the environment by CNE, for the same period, are presented in Table 3.15.8., discriminating between liquid discharges of Tritium and other radionuclides.

Table 3.15.8. - Activity released from CNE to the environment as liquid discharges

YEAR	TRITIUM (TBq)	OTHER RADIONUCLIDES (TBq)
2019	*1.74 x 10^{2}*	*1.84 x 10^{-3}*
2020	*1.23 x 10^{2}*	*1.28 x 10^{-3}*
2021	*1.78 x 10^{2}*	*1.84 x 10^{-3}*

In 2019, 75.6% of the total annual CNE discharges into the environment was tritium and 24.4% noble gases. In 2020, 77.2% was tritium and 22.6% noble gases, while by 2021 94.5% was tritium and only 4.7% noble gases. This was due to the fact that after the restart of the NPP there were some failed fuels.

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 *2019 to 2021*. 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. In spite of 2013 rise, 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)
2019	*6.44×10^{-4}*	*4.35×10^{-3}*	*5.00×10^{-3}*
2020	*5.06×10^{-4}*	*2.98×10^{-3}*	*3.48×10^{-3}*
2021	*1.05×10^{-3}*	*3.95×10^{-3}*	*5.00×10^{-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:

$$\frac{H_p(0,07)}{500\ \mathrm{mSv}} \leq 1$$

$$\frac{H_p(3)}{20\ \mathrm{mSv}} \leq 1$$

$$\frac{H_p(10)}{20\ \mathrm{mSv}} + \sum_j \frac{I_j}{LI_j} \leq 1$$

Where:

H_p (0.07) is the personnel equivalent dose at a depth of 0.07 mm (for skin respectively), integrated in a year,

H_p (3) is the personnel equivalent dose at a depth of 3 mm (for 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 *2019-2021* 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)
2019	*3.8*	*2.9*
2020	*1.7*	*2.1*
2021	*4.2*	*3.0*

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 the *2019- 2021 period* 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)
2019	*0.4*	*0.5*
2020	*2.2*	*1.9*
2021	*0.4*	*0.4*

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 *2019-2021* 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)
2019	*0.4*	*0.4*
2020	*0.3*	*0.4*
2021	*1.6*	*1.1*

The occupational dose in the CNE, *during 2021, was higher than previous years due to the fact that during that year was carried out an Outage for maintenance of the 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 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 - II.

Documentation and procedures for CNA I and CNA II are unified to apply the same methodology in relation to dose optimization criteria.

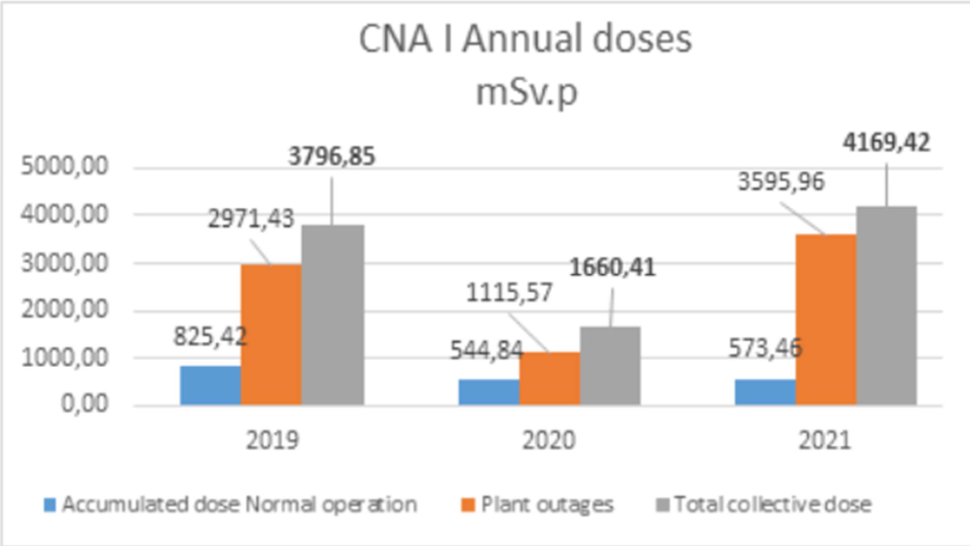


Figure 3.15.6.1

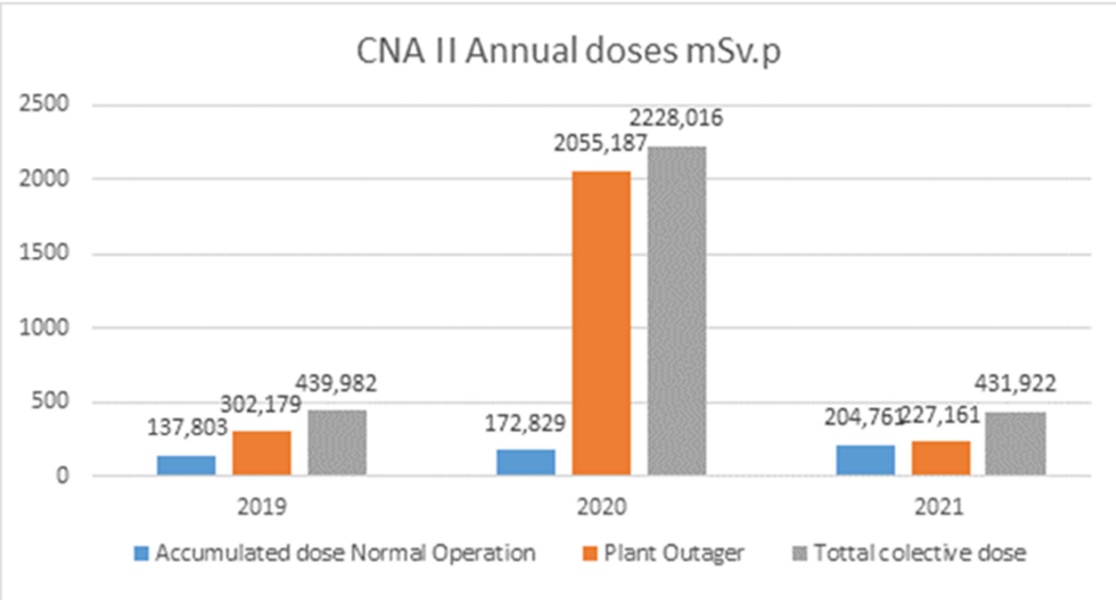


Figure 3.15.6.2

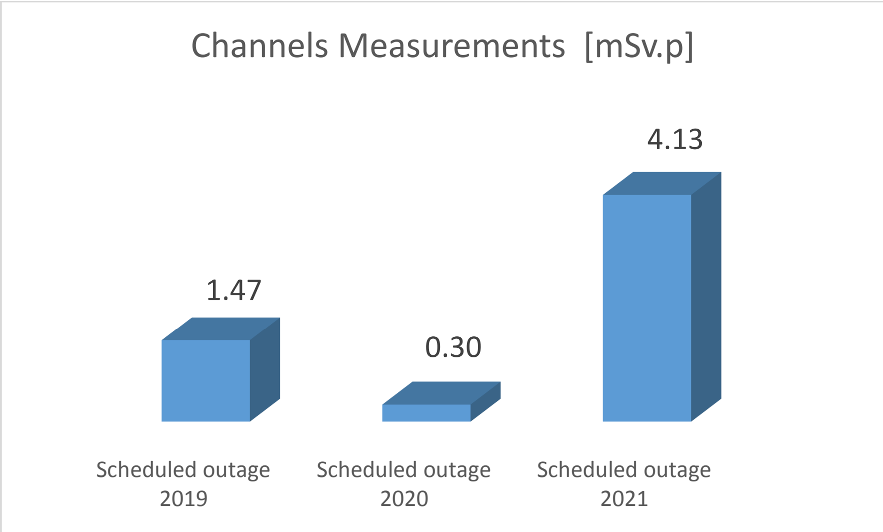
3.15.6.1. CNA I REACTOR INSPECTION PROGRAM

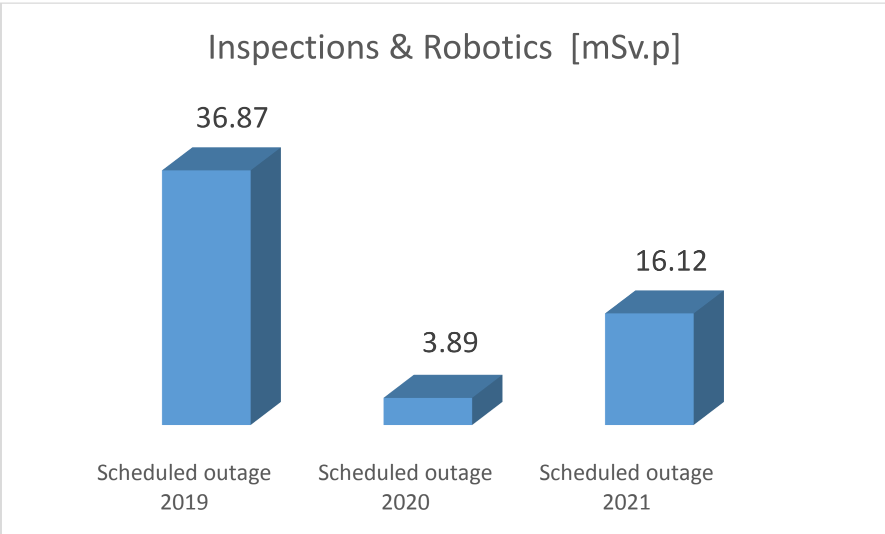
The reactor inspection program requires performing, at each scheduled outage, the metrology of the primary coolant channels (that are transported to the spent fuel storage pool building I) and inspections of the reactor internal components. *To reach the lowest collective dose, it was necessary to:*

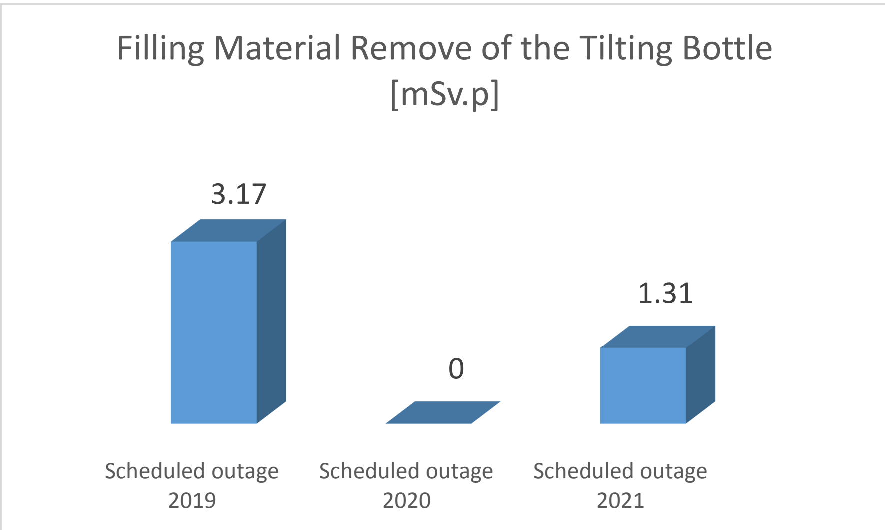
- Shorter inspection times by the use of new tools.
- Increase of control and monitoring of personnel individual doses.
- Optimize staff training for the inspection.
- Reach high degree of decontamination of tools used for inspection.
- Increase use of pressurized suits to remove reactor inspection cameras.
- Improve in shielding of work areas near to the intake of the coolant channels.

Despite the above mentioned, in 2021 it was obtained almost three times the 2019's collective dose. This is because of the number of inspected channels and the required personnel were higher than the corresponding to 2019.

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

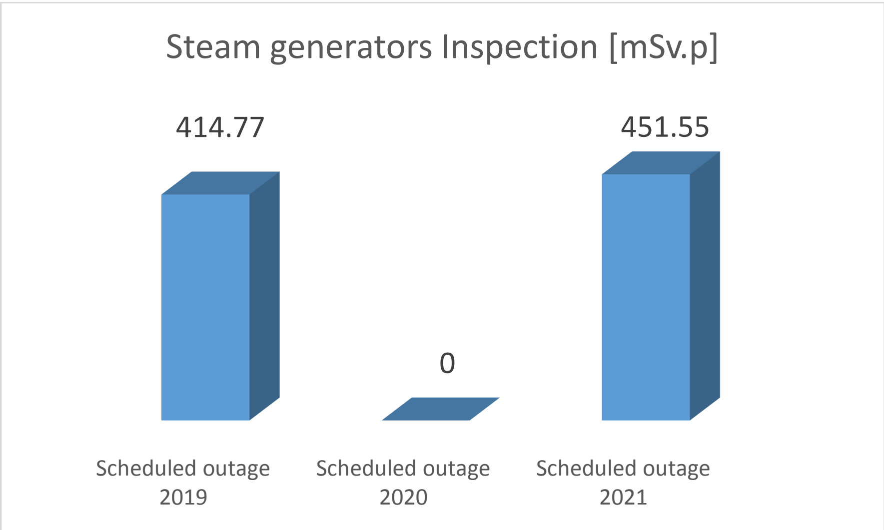
Dose reduction achieved for different scheduled outages due to:

- 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).

In 2019, the steam generator No. 1 was inspected. In general, the inspection of steam generator No. 1 implies more collective dose in comparison with the inspection of steam generator No. 2. The rationale for this is mainly the location, and the radiation fields.

In 2021, the steam generator No. 2 was inspected. Although it was expected less dose than the inspection of No. 1, this was not the case. The reason for this was a reactive change in the scope after one of the bolts used to close the cover of main entrance, was stuck.

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 has extensively worked in reducing doses for CNE staff.

The estimated collective dose for the year 2019 was challenged at the end of it by unplanned outages.

During 2020, hot spots of more than 10 Sievert were detected on two occasions in contact with the refuelling machine, which generated higher collective doses as a result of the removal of these spots.

By applying ALARA's philosophy and criteria, it was possible to achieve the best possible result.

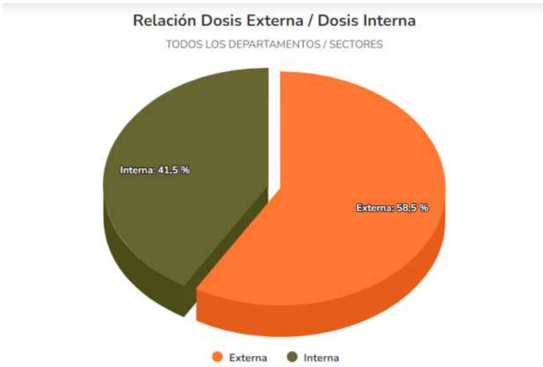
Thus, some of the actions that were carried out were:

- *Management of the ON-LINE Dosimetry System, avoiding unplanned exposures.*
- *Updating of the Digital RWP, reducing the doses of routinely tasks.*
- *Updating of the hot spot database.*
- *Follow-up and trend analysis of the values obtained from the RWP.*
- *Improved signage for rooms and systems.*
- *Staff training and updating.*

All these actions, among others, are part of the 5 years ALARA program. *A significant reduction in the internal vs. external dose ratio can be observed year after year. This is mainly due to the lower concentration of tritium in the air during outages.*

2019

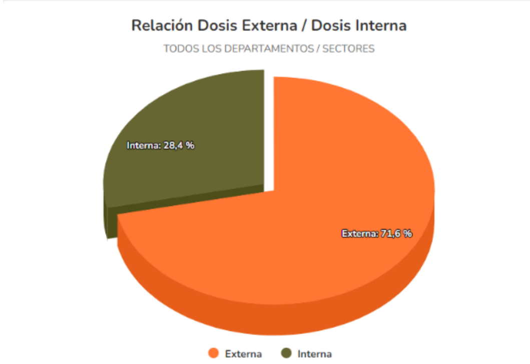
**Estimated (green) vs Actual (Blue) Collective Doses.*



**Internal (brown) vs External (orange) Average Dose*

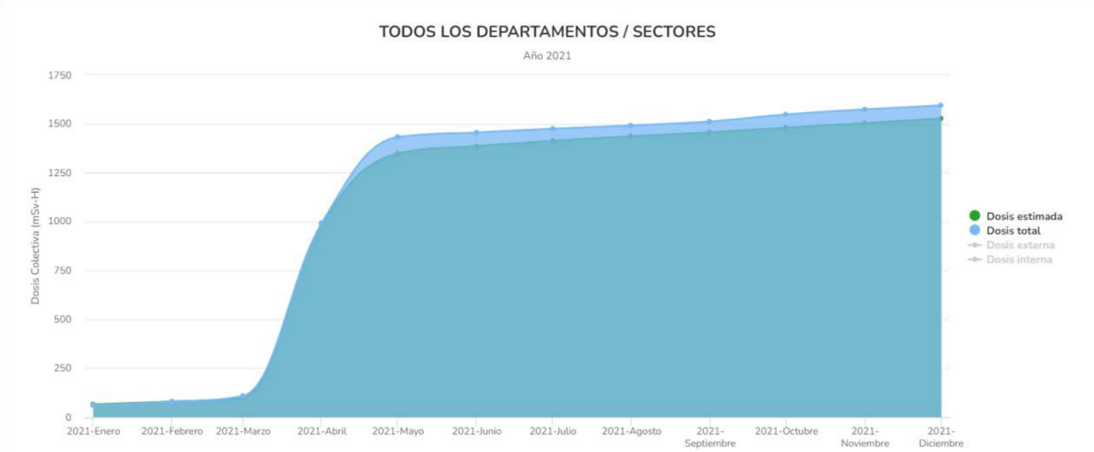
2020

**Estimated (green) vs Actual (Blue) Collective Doses.*

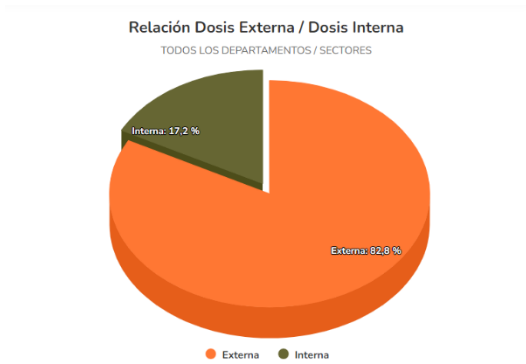


**Internal (brown) vs External (orange) Average Dose*

2021



**Estimated (green) vs Actual (Blue) Collective Doses.*



** Internal (brown) vs External (orange) Average Dose*

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.

Therefore the country complies with the obligations imposed by 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 every aspect related to the necessary strategy to control, mitigate and limit the consequences of possible accidents 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, *municipal and provincial authorities and national 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 NPP´s emergency plans.*
- *ARN verifies that the emergency plans developed by local, provincial and national authorities comply with the regulatory requisites and requirements.*
- *ARN coordinates the representatives of the response organizations regarding the necessary protection actions in case of nuclear emergencies.*

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 *local, provincial and national* 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. *Automatic countermeasures are implemented from the Emergency Operative Center, located near the NPP. In addition, from there the nuclear and radiological evaluation is carried out and the radiological protection of the intervention groups and environmental surveillance are managed, among others.*

The Emergency Operative Center is made up of representatives of all the organizations involved in response (according to what is established in the emergency plans) that belong to civil organizations (Fire Departments, local, provincial and national Civil Protection, etc.), security force (Regional Police, Argentine National Gendarmerie and Argentine Naval Prefecture) and military institutions (Argentine Army, Argentine Navy and Argentine Air Force). These organizations carry out actions so that automatic protection measures are applied in the community.

The Nuclear Emergency Operative Chief (JOEN) is the *Emergency Operative Center 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 function is coordinate 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 Emergency Operative Center and the response strategy is recommended, including radiological and nuclear evaluation, radiological protection of intervention teams, and member of the public 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 *2019)*.

3.16.3. CLASSIFICATION OF EMERGENCY SITUATIONS

Emergencies are classified as following:

- *Alert: Events which lead into an unknown level or a decrease of NPP defense in depth level.*
- *On site emergency: Events which lead into an important decrease of NPP defense in depth level.*
- *General Emergency: Events which lead into a significate risk and require the implementation of off-site urgent protective actions.*

In case of General Emergency, the primary responsible of NPP notifies the local response organizations and ARN. In the last one, the emergency procedures define two stages:

- 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) *and the implementation and preparation of the automatic protection measures of the citizens are requested.*
- Red Alarm: Is declared when the release of 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 requested *in Green Alert.*

During the on-site and off-site emergency, Emergency Operative Center 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 considered under IAEA post-Fukushima recommendations, to extent this zone and include the 360° around the NP P.

Besides to these zones, JOEN considers the following zone for the application of other measures:

- Extended Complementary Planning Zone (EPZ): This zone *covers areas beyond the UPZ zone and* is limited by the results of radiological monitoring *for the implementation of other measures, unlike the measures to be adopted in the PAZ and UPZ. Among them, the instruction to reduce accidental ingestion, restriction of consumption of certain foods, decontamination, etc.*

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 the members of the public receive the pills and then take it at Red Alarm.*

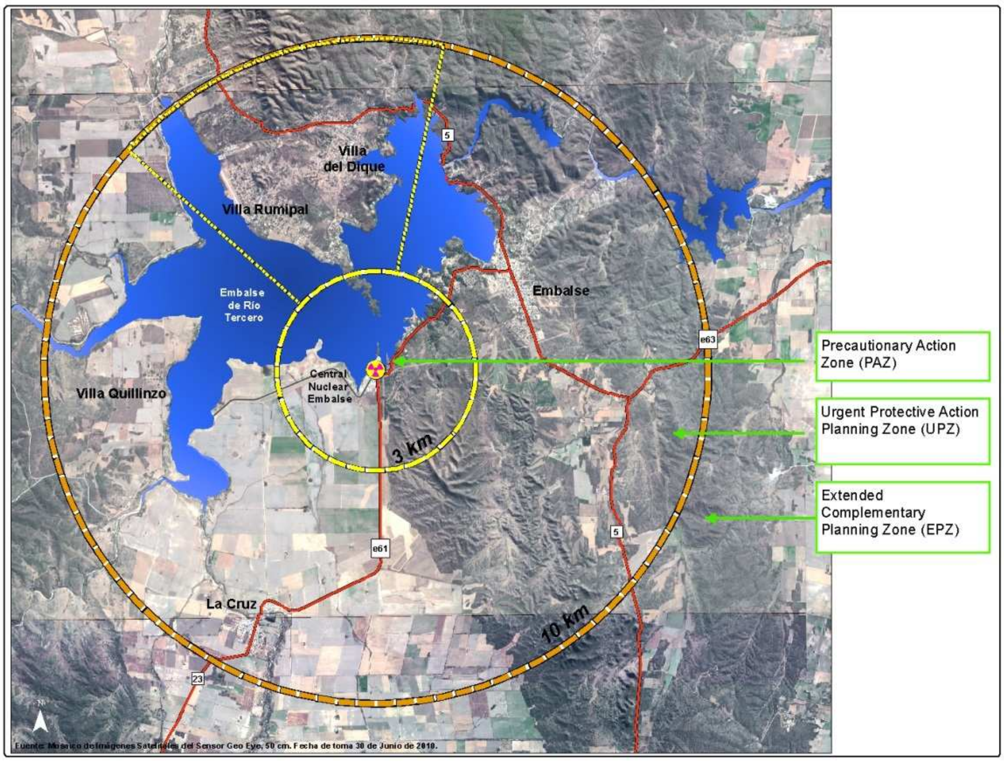


Figure 3.16.1. - Emergency zones map

- Sheltering: It is implemented at UPZ. In Green Alert, the member of the public 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.
- 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, 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 Argentine Food Code and international recommendations (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 response 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.
- 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 drills with the community and response organizations.

3.16.5.1. EMERGENCY PLANS OF RESPONSE ORGANIZATIONS

At present, there are emergency plans in force in the municipalities involved in the predefined emergency zones. In addition, ARN continues to work with the Sub Secretariat for Comprehensive Risk Management and Civil Protection on the new structure of a national plan that covers all areas for the scenario of a nuclear accident.

Otherwise, ARN also works with response organizations in the creation and updating of action protocols for the specific tasks of their concern.

According to the legal framework, the guidelines for emergency plans, protocols and others are established by ARN. In this sense, the plans include agreements and protocols between different organizations.

ARN has 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.

3.16.*5.2.* EMERGENCY PLAN EXERCISES

Each NPP performs the on-site exercise at last once a year. While the offsite drill involving the public and response organizations is carried out alternately once a year between CNE and CNA.

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 response organizations and local community.
- To identify *SWOT (Strengths, Weaknesses, Opportunities and Threats)* and incorporate them as opportunity for improvement.

Off-Site Nuclear emergency exercises performed since last National Nuclear Safety Report, involving population and response organizations, are the following:

- *Atucha NPPs Exercise - Buenos Aires Province, November 14th, 2019.*
- *Embalse NPP Virtual technical meeting due to pandemic, Cordoba Province, December 11th, 2020.*
- *Atucha NPPs Virtual Exercise, Buenos Aires Province, September 30th, 2021.*

The following external organizations, in addition to ARN, participate and receive the specific training for each performed exercise:

- *NA-SA*
- *Sub Secretariat for Comprehensive Risk Management and Civil Protection*
- Argentine Army.
- Argentine Navy.
- Argentine National Gendarmerie.
- Argentine National Prefecture.
- Argentine National Air Force.
- *Argentine Federal Police.*
- Local and Province Police
- Local and Province Civil Defenses.
- Local Fire-fighter Brigades.
- 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 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.

The Security Force assigned for the distribution of iodine pills to population at UPZ is also practiced at the exercise.

Year after year, a greater participation of the population close to the NPP is observed, giving more importance to the Off-Site exercises.

In all emergency exercises, ARN achieved the objective of leading all the organizations response groups.

In addition, as a positive result of response preparedness, the necessary skills are acquired and strengthened to work together with the organizations involved in the emergency.

3.16.6. ARN'S CAPACITIES IN THE PREPARATION FOR EMERGENCIES

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.
- Respond 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.

SIER / SIEN System is activated according to the characteristics of the emergency. Both include 24-hour on-call personnel throughout the year, who are trained and experienced in various fields of action. The System can be activated immediately and operate in such a way that the response is effective and in accordance with the nature and magnitude of the emergency.

SIER / SIEN has different equipment for emergency response, including radiation and pollution measurement equipment, personal protection equipment, logistics and support equipment, among them:

- *High sensitivity equipment to search and rescue lost or stolen sealed radioactive sources*
- *Equipment to air monitoring.*
- *Equipment to detect the presence of radioactive material in major events.*
- *Equipment to carry out environmental control in an accidental situation.*
- *Variety of handheld equipment to detect gamma, beta and neutron radiation, and to measure alpha-beta-gamma contamination, etc.*

On the other hand, in preparation phase, the Emergency Control Center of the ARN uses a software tool to model radiological and nuclear scenarios.

The CCE has a Geographic Information System (GIS) as a complementary software tool. 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.

ARN holds two different consequence evaluation models for nuclear emergencies. 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.

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.

Results obtained from the different models can be integrated to GIS. 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 a communicating system. It includes an independent Local Area Network, which is capable of organize videoconference sessions through the Internet, and an Integrated Digital Service Network. Moreover, a satellite mobile system allows sending information from anywhere on the field to the CCE.

3.16.6.1. MEDICAL RESPONSE FOR EMERGENCIES IN ARN

ARN provides face-to-face and virtual training to different organizations in public and private health institutions about to medical response in radiological and nuclear emergencies. It also collaborates and participates in the internal exercises of medical services of relevant facilities and nuclear power plants.

ARN also made available a free, self-administered video course focused on first responders, especially health personnel. The intervention groups of ARN, Radiological Division of Argentine Federal Police, QBNR Company of Army also contributed.

This course is delivered by medical specialists in radiomedicine, legal medicine, mental health, biological dosimetry, internal and external dosimetry and is the first in Spanish in this mode.

3.16.7. IMPROVEMENTS CONCERNING EMERGENCY PREPAREDNESS IN THE NPPs

Emergency Preparedness is continuously strengthened and improved through the performance of several activities such as audits, external and internal reviews and assessment of applicable international recommendations. In addition, the documentation like emergency plans and procedures, personnel training and equipment availability are kept updated.

The improvements made are summarized below:

3.16.7.1. CNA I AND CNA II

- *There is an Internal Emergency Control Center (CICE) that includes the recommendations and suggestions made by WANO.*
- *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 personnel that make up this organization are part of the plant's personnel.*
- *Since 2019, NA-SA has been working on the general and comprehensive revision of the Site's Emergency Plan, involving the NPP and Headquarters. For its update, the current IAEA documentation was taken into account, as well as the Emergency Plans of other abroad plants. This Plan is currently in the process of being approved by the Nuclear Regulatory Authority (ARN).*
- *There are Emergency Procedures and Severe Accident Guidelines to deal with various situations that may occur at the site:*

  Procedure PS-E-01 "Conformation and functioning of the Emergency Response Organization (ORE)" establishes and describes the organization to face an emergency, defines the roles of the emergency organization positions, alert and notification to the emergency organization, as well as interaction with external response organizations, for example, the Municipal Civil Defense, the Nuclear Emergency Intervention System and the Regulatory Body (ARN).
- *Personnel Training for the ORE:*

  Every year, under a management resolution, an Emergency Exercises Committee is created in which specialists from the different areas of the ORE participate with the aim of designing attractive, challenging and timely scenarios for their own personnel.

Until 2019, four general emergency exercises were carried out per year. According to the regulations, each person involved in the Emergency Response Organization (ORE) must participate in at least one exercise per year. One of those four exercises involved all the site´s personnel.

Starting in 2020, due to the Pandemic situation, different exercises were designed for each sector, group or area of the ORE, avoiding crowds and thus respecting mandatory social distancing. This methodology was very beneficial and efficient, since all the staff has actively participated in each training and/or exercise. Taking into account the results obtained during these years of the pandemic, this exercise methodology will continue to be implemented year after year, incorporating when available exercises with full ORE crews.

In 2020, 50 limited scope Emergency exercises were carried out for small groups and in 2021, 45 were carried out under the same methodology.

Since 2018, exercises with shift changes and exercises to call the on-call ORE crews have been incorporated into the annual planning, where the times of arrival at the Atucha site are measured.

- *The Atucha NPP counts with the "COPERE System". It is a system designed to account for control and online record the people who attend each meeting point during an Emergency Exercise or a real Emergency.*
- *An Emergency Call System was designed, which objective is to perform calls to the Atucha NPP on-call personnel available at the time of the emergency situation, automatically through pre-recorded telephone calls. The average call time for all the on-call personnel is 20 minutes.*
- *There is an Emergency Equipment System (EN), which allows keeping a record, control and monitoring of the availability of the Emergency Equipment defined by the plant.*
- *Every year, the Atucha NPP communicates the Emergency Plan in all the educational centers that are located within the UPZ, as well as in the External Emergency Response Organizations (OREE).*

3.16.7.2. CNE

The actions carried out are mentioned below:

- *An Internal Emergency Control Center (CICE) was prepared considering the recommendations and suggestions provided by WANO.*
- *The Emergency Response Organization (ORE) was made up of 81 positions, each of which is made up of a minimum of 3 to 6 CNE site staff.*
- *Since 2020, CNE and NA-SA Headquarters have been carried out a comprehensive review of the Site Emergency Plan. For its update, the current IAEA documentation was taken into account, as well as the Emergency Plans of abroad plants. Currently said document is under the approval process of the Regulatory Body (ARN).*
- *Emergency Procedures and Severe Accident Guidelines were developed to deal with various situations that may occur on site.*
- *Procedure PS-101 for "Conformation of the Emergency Response Organization (ORE)" and procedure PS-102 for "Functioning of the Emergency Response Organization (ORE)" establish and describe the organization to deal with an emergency. They define the roles of the positions of the emergency organization, the alert and notification to the emergency organization, as well as the interaction with external response organizations, for example, the Municipal Civil Defense, the Nuclear Intervention System of Emergency and the Regulatory Body (ARN).*
- *ORE Staff Training:*

  Every year, an annual schedule of training, drills and exercises is carried out, which includes the programming and objectives for each exercise with the goal of designing realistic, challenging and timely scenarios for its own personnel. The training sessions are divided into partial and general emergency drills and exercises. According to the current regulations, each person involved in the Emergency Response Organization (ORE) must participate in at least one exercise per year.

In 2020 and 2021, due to the Pandemic situation, different partial exercises were designed for each sector, group or area of the ORE, avoiding the crowding of people and thus respecting mandatory social distancing, suspending general exercises.

In 2020, 57 limited scope exercises were carried out for small groups and in 2021, 49 exercises were carried out under the same methodology.

Since 2018, exercises with shift changes and exercises to call the available shifts of the ORE have been incorporated into the annual planning, where the times of arrival at the CNE are measured.

An alternative Emergency Call System was designed. The objective is to carry out communications calling the personnel on duty so that they assist the CNE at the time of the emergency. This is done automatically through pre-recorded phone calls. This system is intended to provide redundancy to the Main Call System for ORE personnel by means of a beeper.

An Emergency Equipment System (EE) allows for a record, control and monitoring of the availability of the Emergency Equipment defined by the CNE. In addition, all equipment has its identification and its associated preventive maintenance.

Every year, the CNE disseminates the Emergency Plan in all the educational centers that are located within the UPZ, as well as in the External Emergency Response Organizations (OREE).

3.16.7.3. EMERGENCY COMMITTEE HQ

Since 2018, it was created by Resolution an Emergency Committee at the Headquarters level to mitigate plant damage and support response actions.

Procedure NA-02-13 "Conformation of the Emergency Committee" establishes and describes the organization, the roles of the positions of the emergency Committee, the alert and notification to the emergency organization.

The committee is made up of 22 positions and each position is staffed with at least two people.

Exercises are performed for specific areas and for the role holder with general training courses for all the roles. Drills and Exercises are developed with CNA and CNE's ORE.

CCE was defined to be operative for the response of the Emergency Committee and the specific equipments are tested and maintained functionality ready to respond.

3.16.8. SIER/SIEN DURING PANDEMIC OF COVID-19

During the social isolation due to COVID-19, the SIER/SIEN radiological and nuclear emergency intervention systems remained active, assuring the operation of the response system. The pandemic has not affected the operation of the response groups at any time.

Maintenance was also carried out on all the equipment and devices necessary for the normal development of the CCE's tasks (communications system, multimedia, radiation detectors, surface contamination detectors, vehicles, etc.).

SIER/SIEN activities continued to be carried out virtually, such as the evaluation of NPP emergency plans, training for first responders, technical meetings (attendance at webinars, workshops of the IAEA and other organizations), dissemination of the NPP emergency plan, among others.

According to the Nuclear Activity Act No. 24,804, an external exercise of the nuclear emergency plan must be carried out. Despite the pandemic situation, SIER/SIEN has carried out a desktop exercise virtually with the participation of all the response agencies involved in accordance with the provisions of the NPP´s emergency plan.

During the pandemic, the training has also continued in the Hospitals of the Armed Forces, together with Armed Forces Well-being and Health Coordination under Ministry of Security.

Virtual courses were held for the health systems of the different levels of care, around the Atucha site.

3.16.9. 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.

Within the framework of the Convention on Early Notification of Nuclear Accidents, representatives of Uruguay and Brazil have been invited as observers in some of the exercises carried out in Argentina.

3.16.10. 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.

Preparation activities for emergencies related to nuclear safety in NPPs continue to be developed as planned.

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 fulfils all the obligations imposed by 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 as part of the Preliminary Safety Analysis Report and must be revised if it is necessary according to the results of the Periodic Safety Review (PSR), for the Operating License renewal.

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” was issued 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 was finished in 2019 and the plant is currently under normal operation.

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 Argentine Republic), 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 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 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, 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 there were weaknesses or critical situations (cliff edge effects) found, which would make necessary to take urgent actions.

The improvements and modifications proposed by the NPPs Licensee 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.

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, is described in the following sub-sections.

3.17.2.3.2.1. CNA I and CNA II site re-evaluation

For Atucha site, 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 *was* carried out with the objective 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 *was* 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 was derived from the mean uniform hazard response spectra (UHRS) for a recurrence period of 10,000 years.

The CNA I NPP, *consistent with the criteria and requirements established in the 1960s for nuclear power plants located at sites of low seismicity,* 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 as 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 consisted 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. *They* have already been solved *during this period*.
- 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. In addition, to fulfill this commitment, replacement relays have been purchased and CNEA has been contracted to perform qualification tests. It is expected to be completed during Life Extension.
- Category 3 corresponds to items dependent on both easy fixes and a relay chatter evaluation. *Easy fixes have already been solved during this period. Regarding relay chatter, it will be completed during Life Extension (see category 2).*
- Category 4 corresponds to items for which HCLPF calculations were performed. *During this period, many HCLPF capacity calculations were performed.*
- Category 5 corresponds to items to be evaluated in Phase 5 of the seismic safety evaluation programme. They correspond to the following:
    - Structural capacity and safety margin of the Second Heat Sink Building *(finished)*;
    - Masonry Walls (interaction effects) *(finished)*;
    - Refuelling Machine and Tilting Bottle *(finished)*;
    - NSSS seismic capacity evaluation *(under process)*;
    - Components of the Moderator System (MOV FIAT valves, heat exchangers, vertical pumps) *(under process)*.
- 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.
- Installation of supports.

Seismic instrumentation:

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.

Liquid levels released per site and potential flood levels were calculated. The affected safety components were identified and actions were taken, such as:

- *Design, construction and installation of non-return mechanisms in different enclosures.*
- *Replacement of normal doors with watertight doors.*
- *Placement of qualified seals (flame retardant, fireproof, waterproof and radiation resistant) in certain slab and wall passages.*
- *Placement of level sensors.*
- *Reconditioning and/or replacement of emergency door (to provide watertight sealing) that connects Level -6.50 m of Auxiliary Building with Manoeuvre Building, to isolate the controlled zone from floods that come from outside.*

CNA II

The SSA consisted 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. Almost all easy fixes have already been solved.

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 operates is -0.5 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 by INA (National Water Institute).

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:

- For maximum level rise: a chain break of Itaipú and Yacyretá dams, 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 event's duration of 24 and 48 hr.
- 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.

On the other hand, the report from INA took into account the study "Risk analysis of flood duration in the coastal areas of the Río de la Plata considering climate change". This study models the influence of the Meteorological event called sudestada (strong winds from southeast) over the La Plata River levels and Paraná River regime. The implications of this model can be extrapolated to recurrence periods of 1,000 years.

The impact of climate change was also considered, where possible scenarios for 2030 and 2070 are modelled, predicting a major influence of south-east winds (higher levels and longer duration). It is worth mentioning that CNA already makes a continuous tracing and record of Paraná River levels. These data can be checked out with the data released by governmental services, like National Naval Prefecture.

During 2020 and 2021, due to La Niña phenomenon, an extraordinary low water level in the Paraná River took place. New on-site monitoring system on the river was installed and reassessment about pumps operating was carried out. Work was done with INA and specific procedures were developed to monitor the level of the river.

Additionally, a hypothetical scenario involving an earthquake strong enough to break down Yacyretá dam and causing, at the same time, a seismic movement at CNA 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 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 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 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). *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 following guidelines *and internal instructions* were drawn up within the framework of severe accident management:

- Guide *2-GAS-CE- 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 *2-GAS-CE 05* "Supply and Venting of Steam Generators".
- *ORE-015: Water supply to the tanks of the GHC system - Demineralized water from the SGA fire system- Fire network.*
- *ORE-007: Power supply SGA system from the fire network of the UG-PN workshop.*

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.

In relation to aircraft traffic, according to IAEA Standard NS-G-3.1 "External Human Induced Events in Site Evaluation for Nuclear Power Plants" (2002), and the information provided by the Argentine Air Force, both airways located over CNA as the proximity of airports can be ruled out as sources of risk since they are beyond the Screening Distance Value (SDV) recommended for airways and proximity to airports (4 km). In addition, the CNA zone is a prohibited flight zone because the restriction is 3000 feet (914.4 m) and the airways are above that height. The annual probabilities of a plane and helicopter accident in the CNA linear airway are less than 10^{-7}/ year, so according to the Argentine Regulatory Guide AR 3.1.3. these scenarios are discarded.

Related to potentially hazardous industrial plants, according to the information provided by the Municipality of Zárate, within this area there are chemical factories, and few of them are dangerous for human health and with the possibility of forming a toxic cloud. All of them are beyond the SDV suggested by the IAEA for explosions and toxic clouds.

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, related to 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's 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 was performed 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.

The ARN assessed the results of the stress tests concluding that the approach used by the Licensee for the SSA re-evaluation was appropriate. In addition, ARN and IDIA as regulatory body's technical support organization (TSO) performed a review of the site seismic hazard assessment conducted by the Licensee. The results were the basis for ARN to define the following regulatory requisites for CNE life extension project:

- New SSCs´s design must be done considering, as design basis, the load obtained from the updated UHS;
- 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; and
- 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 ARN.

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.

<u>Operating procedure for post-earthquake actions:</u>

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.

<u>Addition of a seismically qualified rupture disc assembly to the existing inspection port of the calandria vault:</u>

A 24-inch rupture disc (RD) assembly was 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.

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 Embalse Río Tercero dam and the consequences of different combinations of dam’s breaks and weather conditions on CNE.
- An air compressor driven by a DG located in the turbine building has been 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 center 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 ARN carried out the assessment of the stress tests results and concluded the following:

- There were identified not relevant weaknesses requiring non-urgent actions.
- It has been verified 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 proposed 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 and addressed.

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 Embalse 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.

As a result of the assessment of risk of tornadoes for the CNE site, the following actions were performed:

- *Cleanup of outdoor areas, all loose/stored components that could lead to be potential missiles.*
- *In order to guarantee the 5 key safety functions under the occurrence of a tornado, an assessment was performed and it was concluded the need for reinforcement measures in civil structures in EPS and EWS buildings and the housing where some containment isolation valves are located.*

The consideration of tornadoes and wind loads for CNE is consistent with both domestic regulations and international criteria, established at the design time. However, it was re-evaluated the risk of tornadoes for the site and there weren´t identified nor the need for additional provisions neither for protective measures.

In addition, lightning and intense rains have been analysed and it is considered that the licensee carried out the appropriate actions to successfully cope these scenarios.

3.17.3. IMPACT OF THE INSTALLATIONS ON INDIVIDUALS, SOCIETY AND ENVIRONMENT

The operating restrictions 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 Argentinian NPPs operator disposes of Operating Procedures for Abnormal Events (POEAs) foreseen to manage abnormal events and mitigate the corresponding consequences.

The development of Severe Accident Management Program (SAMP) was continued at CNA I-II and CNE sites.

The Environmental Management System (EMS) implemented by the Licensee adopts the requirements of ISO *14001:2015* “Environmental Management Systems-Requirements with guidance for use” (see section 3.13). 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 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.

3.17.4. 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.

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.

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.

Criteria of the Regulatory Standard AR 3.2.1. 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.

Criteria of 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 by 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. ARN authorized the transfer process of 1477 CNA I fuel elements to CNA II pools. The fuel transfer requirements are average burned lower than 6740 MWd / tU and minimum decay time of 33.5 years. This process finished in May 2019, with 1435 spent fuel elements transferred. In addition, a rearrangement of spent fuel elements between the pools was 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).
- *New level and temperature measurement was installed to manage water replenishment to the spent fuel pools.*

The following design measures / changes are still under assessment:

- *Corium-Barrier to improve Containment long term integrity.*
- I&C improvements to provide the information for severe accidents management (beyond design basis accident).

Besides, in the frame of LTO, a comparison of the current design against the latest KTA standards *was carried out and reviewed by the ARN*. 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 and regulatory requirements, 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.

As it was mentioned, 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, QSP 4a and 15a/c Rev. 2 mentioned in PSAR of CNA II).

The QAP 115 program ended when the CNA 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 assessment 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 behaviour 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).
- *Alternative power sources (MDG).*
- *New level and temperature measurement was installed to manage water replenishment to the spent fuel pools.*

The following design measure / change is still under assessment:

- *Corium-Barrier to improve Containment long term integrity.*

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 280. The stored inventory at the end of 2021 was 131,760 fuel elements in 244 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.

- Improvement of the seismic capacity of the whole plant to cope with the updated DBE.
- *Alternative MSSVs actuator air injections for remote manual opening.*

CNE has been working on the implementations of the following improvements since 2019:

- *Long range containment activity monitor.*
- *Containment hydrogen measurements systems.*
- *Replacement of the emergency diesel instrument air compressor.*
- *Backup nitrogen make up to the dousing tank spry valves.*
- *Primary side ECC heat exchanger bypass.*
- *Duplication of the containment sump filters of the medium and low pressure ECC pumps.*

3.18.3.4. CAREM PROTOTYPE REACTOR

The concept of CAREM *prototype reactor* 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 prototype reactor 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 *IV*.

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.

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 *2026*, 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 *Eighth* 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 is 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

The licensing of CAREM prototype reactor has introduced the issue to ascertain whether changes to the existing regulatory framework could be necessary for future small modular reactors. Following this, licensing small modular reactors is a learning process for ARN.

For the particular case of CAREM a licensing model applicable to the construction and commissioning stages was defined. This model established the milestones for the beginning of construction and for granting the construction authorization. In particular, the ARN established requirements for additional mandatory documentation with respect to the traditional nuclear power plant licensing model. This authorization license model was approved by the ARN in 2010 and was formally communicated to the Responsible Entity (CNEA).

The beginning of construction of the reactor and others regulated activities, were authorized by ARN with the issue of "Authorization to Use the Site and Construction (AUSC)". The mandatory documentation required for obtaining such authorization consisted in the submission of the following subjects:

- *Design Information (PSAR format).*
- *Environmental Radiological Impact.*
- *Waste Management.*
- *Radiological Emergency.*
- *Quality management.*
- *Project Schedule.*

The above mentioned authorization contains as "license conditions" a Regulatory Requirement (RQ-CAREM-003) strengthening some items of the authorization. In particular, this Regulatory Requirement (RQ) deals with some findings from the PSAR review and assessment (related to civil work, either structural, confinement or shielding issues) and its fulfilment conditioned the beginning of the construction of the nuclear module of the reactor.

In December of 2014, CNEA presented (in accordance to ARN's expectations) the documentation containing the corrective action and thus, ARN authorized the initiation of nuclear module construction activities.

Between 2016 and 2018, the development of the civil works of the project was slowed down due to the change of CNEA's contractor to carry out the work. The ARN participated in the control of the transition process to the new civil works contractor.

In 2019, the ARN decided an update of the licensing model for facing the following stage in the lifecycle; Commissioning. This decision was driven by the evolution of the CAREM project and the experience gained in other licensing projects (Atucha II NPP, Embalse NPP Long-Term Operation). The selected model is in compliance, as far as practicable, with the existing licensing process for NPPs in terms of mandatory documents (table of contents and scope) and general approach. This was formally communicated to CNEA in October 2019.

In terms of regulatory process, the next licensing milestone will be the issuance of a Commissioning License by which ARN will authorize the performance of tests in accordance to a programme that has to be timely submitted for the regulatory review and approval.

3.18.3.4.4. Design and Engineering Tasks

Defense in Depth (DiD) Principle is the basis for the process of safety internalization in the design. Appropriate criteria were defined for DiD internalization in the design taking into account CAREM prototype reactor general design characteristics, to properly and in a balanced way prevent, control and mitigate the postulated events.

Western European Nuclear Regulators Association proposal for DiD levels definition was adopted, which allows for the consideration of Multiple Failures Events (Level 3B) as part of DiD Level 3 for the prevention of core damage. A strategy was defined for each one of the levels, considering two stages for level 3A and *level 3B. For level 3A, the first stage was accomplished by means of passive safety systems while the second stage was implemented by active systems in order to achieve the final safe state. For level 3B, the first stage was also accomplished by systems with passive processes while the second stage by simple systems with external water supply.*

Subsequently, starting from the Fundamental Safety Functions (FSFs) and by means of attributes Low Level Safety Functions (LLSFs) including monitoring ones were identified for each one of the DiD levels and stages.

Safety Functional Groups (SFGs) -set of structures, systems and components (SSCs) that fulfill a function- were defined for each one of the LLSFs. Afterward, safety classification process was executed. The methodology developed is in harmonized with the International Electrotechnical Commission and the IAEA was applied. Criteria for safety categories allocation to LLSFs and classes to SFGs/SSCs were established, based on the strategy defined to internalize DiD in CAREM 25 design. Three categories for LLSFs and three classes for SFGs/SSCs were defined, considering appropriate rules for class reduction that were applied, for example, to SSC that constitute the Diverse Line of Protection (Level 3B). As part of this process, deterministic and probabilistic evaluations were also considered to support the evaluation of the relative importance of SSCs.

Finally, and by setting deterministic and probabilistic design acceptance criteria for each DiD Level, systems were evaluated to verify that the assigned functions were properly fulfilled. In a complementary way and by means of Probabilistic Safety Assessments and Deterministic Safety Analysis, design feedback was provided for DiD Levels 2 and 3 systems.

The Chemical and Volume Control System was proposed to control Loss of Heat Sink event and the whole postulated spectrum of LOCAs within DiD Level 2; a Reactor Pressure Vessel (RPV) depressurization system and others are also proposed to cope with Multiple Failures Events (Level 3B). A pyramidal set of technical documentation was implemented and IAEA safety standards were taken as reference to carry out the whole process.

It is concluded that this approach, based on a consistent DiD implementation, has provided a clear roadmap to support the design of systems important to safety and their balanced integration into the plant. This has considerably facilitated the engineering development and the regulatory licensing processes of CAREM prototype reactor.

3.18.3.4.5. Construction Tasks

As it was mentioned, the civil works began after obtaining the necessaries authorizations and in February 2014, the first structural concrete was poured out*, and the first nuclear concrete was poured in August 2015.*

The *construction of the conventional building including the turbine, the generator and the tertiary system has already begun.*

In the period 2019 - 2022, the progress of the civil works of the project was delayed due to the restrictions established by the global pandemic of COVID-19.

Once the restrictions were lifted, the regulatory inspections at the site returned to their normal development in accordance with the inspection program for the construction of civil structures. It is important to mention that this program was developed by ARN inspectors in conjunction with experts belonging to the main TSO in civil area and it is aimed at addressing the results from the review of the mandatory documentation.

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.

On February 1st, 2022, NA-SA and China National Nuclear Corporation (CNNC) signed an Engineering, Procurement and Construction contract (EPS) for the fourth NPP Project, which will be a HPR-1000 (Hualong I) designed by 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 meets the requirements of current 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.

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.*

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.

The licensing activities have not been initiated at the time of developing this National Report.

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 by 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, *Technical Operating Specifications* and the Policies and Principles Manual.

ARN inspectors verify that the Mandatory Documentation that includes the 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, Design Authority, 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 Preliminary Testing Program (pre-nuclear) and an Organization to carry it out. The preliminary test program comprised those tests required to demonstrate that the as built plant fulfil the design intention as described in the technical specifications.

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 Regulatory Body 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 commissioning 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.

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 Technical Specifications 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 *with an important review of systems includes, components and basic normative, drawing a parallel, when it may possible, with ISI program of CNA II.*

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”.

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 *Reactor* Internal Component Surveillance Program *is routinely carried out both in CNA I and CNA II*, 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 and 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.

As was mentioned in Section 3.19.3., 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.

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 CANDU Energy (former 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.

The most significant operational events in *CNA (*CNA I *and* 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 (*CNA I *and* 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.
- *Identified lessons learned from the analysis.*
- Taking immediate corrective actions to avoid event occurrence or recurrence.
- Corrective action follow-up.

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 *and operations* 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 regard,* 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 (*CNA I *and* 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 (*CNA I *and* 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 (*CNA I *and* 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 (*CNA I *and* 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 *(Unit 1), Cernavoda (Unit 1 and 2), Qinshan (Unit III-1 and III-2), and Wolsong (Unit 1-4),* in order to exchange operating experience. Moreover, it is member of the CANDU Owners Group since its creation.

Presently, both *CNA (*CNA I *and* CNA 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 *and CNA II since 2014*. 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*,* *Procurement, Construction and Commissioning* phases of the Life Extension Project. Nevertheless, there is still some useful Operating Experience that could be implemented *after* Commissioning Phase *as some events could uncover hidden issues which were originated during Life Extension Project.* For the period *2019 - 2022,* it could be mentioned the following:

- The Headquarters Operating Experience Division has selected *135* reports from international databases.
- The CNE Operating Experience Division has received *117* reports. Approximately *10%* of them are “Significant” and the rest are “For information”.
- The CNE Operating Experience Division sent *77* responses to the events reported, including *70* proposals of corrective actions.

Examples of lessons learned from local and international operating experience in the period *2019* - *2022* are shown in Annex V.

3.19.8.3. PEER REVIEWS AND ACTIVITIES BETWEEN THE LICENSEE AND WANO FROM 2019 TO 2022

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), Crew Performance Observations (CPO), Corporate Peer Review (CPR) and Follow-Up (FU) missions from 2019 to 2022:

- *CNA I-II: FU in 2019 (of PR 2017), Crew Performance Observations (CPO) and Design Informed PR in 2021.*
- *CNE: Crew Performance Observations (CPO) and PR in 2021.*
- *Corporate: Corporate Peer Review will be done in 2022.*

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 *also provides* specialists from the NPPs to participate in every WANO Program.

Some of the most important activities *carried out* during the reported period are listed in Table 3.19.1.

Table 3.19.1.

PERIOD	HOST	PARTICIPANTS	ACTIVITY
July 11, 2019	*Atucha NPP*		*Exit Meeting Peer Review Follow Up*
July 22-26, 2019	*Atucha NPP Embalse NPP NA-SA Corporate*		*WANO Representative Visit*
October 2-4, 2019	*Embalse NPP*		*Peer Review Pre Visit*
October 10, 2019	*Embalse NPP*		*Annual Meeting*
October 20-23, 2019	*London, United Kingdom*	*NA-SA President*	*Biennial General Meeting*
October 20-23, 2019	*London, United Kingdom*	*NA-SA Governor*	*Biennial General Meeting*
November 14, 2019	*NA-SA Corporate*		*Corporate Peer Review Kick Off Meeting*
November 19-21, 2019	*Heysham NPP United Kingdom*	*Atucha NPP Maintenance Deputy Manager*	*Maintenance Fundamentals Workshop*
December 2-6, 2019	*Embalse NPP*		*SOER Recommendations Support Visit*

February 12-13, 2020	*Videoconference with Atucha NPP, Embalse NPP and NA-SA Corporate*		*WANO Representative Visit*
June 10, 2020	*Videoconference*	*Atucha Site Manager*	*COVID-19 Human Performance Forum*
June 10, 2020	*Videoconference*	*Embalse NPP Plant Manager*	*COVID-19 Human Performance Forum*
June 23-24, 2020	*Videoconference with Atucha NPP, Embalse NPP and NA-SA Corporate*		*WANO Representative Visit*
June 24, 2020	*Videoconference*	*NA-SA Governor*	*76th WANO Paris Centre Governing Board*
September 30, 2020	*Videoconference*	*Atucha Site Manager*	*Site Vice President & Plant Managers' Conference: Leadership in Nuclear Plant Performance*
September 30, 2020	*Videoconference*	*Embalse NPP Plant Manager*	*Site Vice President & Plant Managers' Conference: Leadership in Nuclear Plant Performance*
October 12-14, 2020	*Videoconference*	*NA-SA´s WANO Interface Officer*	*69th WANO Interface Officer Meeting*
October 28, 2020	*Videoconference*	*NA-SA Alternate Governor*	*81st WANO Paris Centre Governing Board*
November 5-6, 2020	*Videoconference with Atucha NPP, Embalse NPP and NA-SA Corporate*		*WANO Representative Visit*
November 16-20, 2020	*NA-SA Headquarters*		*Corporate Peer Review Pre-Visit*
December 9, 2020	*Videoconference with Atucha NPP and Embalse NPP*		*WANO Representative Visit*
December 10, 2020	*Videoconference*		*Annual Meeting*
March 5-11, 2021	*Videoconference Atucha NPP*		*Peer Review Pre-Visit*
March 11, 2021	*Videoconference*	*NA-SA's Governor*	*82nd WANO Paris Centre Governing Board*
March 12, 2021	*Videoconference*	*NA-SA's Chief Nuclear Officer*	*Chief Nuclear Officer Forum*
March 25, 2021	*Videoconference*	*Embalse NPP Technical Assistant to Plant Manager*	*WANO Representative Support Visit Training needs identification and management of training activities*
March 26, 2021	*Videoconference*	*Embalse NPP Technical Assistant to Plant Manager*	*WANO Representative Support Visit Peer Review Preparation*

March 30, 2021	*Videoconference*	*Atucha NPP Head of Site Manager Technical Assistance Department*	*WANO Representative Support Visit Leadership and Risk Management*
March 30, 2021	*Videoconference*	*Atucha NPP Head of Human Factors Department*	*WANO Representative Support Visit Leadership and Risk Management*
March 30, 2021	*Videoconference*	*Atucha NPP Head of Engineering Technical Support Division*	*WANO Representative Support Visit Leadership and Risk Management*
March 30, 2021	*Videoconference*	*Atucha NPP Head of Operating Experience Section*	*WANO Representative Support Visit Leadership and Risk Management*
March 31, 2021	*Videoconference*	*Atucha NPP Head of Engineering Technical Support Division*	*WANO Representative Support Visit Radiological Safety and Peer Review Preparation*
March 31, 2021	*Videoconference*	*Atucha NPP Head of Radioprotection Department*	*WANO Representative Support Visit Radiological Safety and Peer Review Preparation*
April 26-27, 2021	*Videoconference*		*Corporate Peer Review Training Seminar*
May 31-June 4, 2021	*Atucha NPP, Argentina*	*Atucha NPP Head of Shift Operations Division*	*Crew Performance Observations (Host Peer)*
June 7-25, 2021	*Embalse NPP, Argentina*		*Peer Review*
June 7-25, 2021	*Embalse NPP, Argentina*	*Embalse NPP Professional in training Safety and Radioprotection Department*	*Peer Review (Host Peer)*
June 24, 2021	*Videoconference*	*NA-SA´s Alternate Governor*	*83rd WANO Paris Centre Governing Board*
June 24, 2021	*Videoconference*	*NA-SA´s WANO Interface Officer*	*83rd WANO Paris Centre Governing Board*
August 30-31, 2021	*Videoconference*		*Stream Analysis Embalse Peer Review*
September 14-16, 2021	*Videoconference*	*NA-SA´s WANO Interface Officer*	*72nd WANO Interface Officer Meeting*
September 14, 2021	*Videoconference*		*Improvement Action Plan Meeting Embalse Peer Review*
September 17, 2021	*Videoconference*		*Exit Meeting Embalse Peer Review*
October 13, 2021	*Videoconference*	*NA-SA´s Governor*	*CEO Meeting*
October 13, 2021	*Videoconference*	*NA-SA´s Governor*	*84th WANO Paris Centre Governing Board*

October 14, 2021	Videoconference		Annual Meeting
November 3-19, 2021	Atucha NPP, Argentina		Peer Review
November 3-19, 2021	Atucha NPP, Argentina	Embalse NPP Chemical Engineering	Peer Review (CY Host Peer)
November 3-19, 2021	Atucha NPP, Argentina	Atucha NPP Head of Industrial Safety Division	Peer Review (MA Host Peer)
November 15-17, 2021	Videoconference	NA-SA Corporate Lead Evaluator Independent Oversight Embalse NPP	Industry Working Group Independent Nuclear Safety Oversight
November 17, 2021	Atucha NPP	NA-SA Operative General Manager	WANO Paris Centre Director Meeting
November 22-December 2, 2021	Videoconference	NA-SA Corporate Head of Internal Communication and Visual Identity Division	Corporate Peer Review (Communications Reviewer)
November 29-December 3, 2021	Embalse NPP, Argentina		Crew Performance Observations
November 29-December 3, 2021	Embalse NPP, Argentina	Embalse NPP Head of Operations Assistance Department	Crew Performance Observations
December 15, 2021	Atucha NPP, Hybrid		Stream Analysis Peer Review
December 16, 2021	Atucha NPP, Hybrid		Improvement Plan Peer Review
February 28-March 4, 2022	Videoconference	NA-SA´s WANO Interface Officer	73rd WANO Interface Officer Meeting
March 15, 2022	Videoconference	NA-SA's Governor	85th WANO Paris Centre Governing Board
April 12-14, 2022	Videoconference	NA-SA Nuclear Project Management Project Deputy Manager – Atucha III Project	Industry Working Group New Unit Assistance
April 12-14, 2022	Videoconference	NA-SA Nuclear Project Management Technical Deputy Manager - Atucha III Project	Industry Working Group New Unit Assistance

The following activities will be performed in NA-SA during the rest of *2022:*

- *Chief Nuclear Officer Forum - Virtual (April 27, 2022)*
- *WANO Interface Officer Meeting:*
    - *74th WANO Interface Officer Meeting – Virtual (May 23-25, 2022)*
    - *75th WANO Interface Officer Meeting in Eletronuclear, Brazil (September 18-23, 2022)*
- *Governing Board:*
    - *86th WANO Paris Centre Governing Board in Swissnuclear, Switzerland (June 16, 2022)*

    - *87th WANO Paris Centre Governing Board in Czech Republic (October 10, 2022)*
- *Biennial General Meeting in Czech Republic (October 9-11, 2022)*
- *Corporate Peer Review in NA-SA Headquarters (April 18-28, 2022)*
- *Annual Meeting in NA-SA Headquarter (date to be defined)*

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 *Seventh* National Report to the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management (JCSSFMSRW) *(2020)* 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 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 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)".

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. 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 *by the end of 2022.*

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:

- Reorganization of the reactor's internal components placed in the decay pools hangers within the Pool Building I and II.
- 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 current CNE's spent fuel dry storage system (ASECQ) has *280* silos. The stored inventory at the end of *2021* was *131,760* fuel elements in *244* 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. *During 2022, some SF will be repositioned within the storage pools and the hanger from the Unit I will not be used any more. In that moment all the spent fuel storage pools will have 1,512 spent fuels capacity.*

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 is used to transport it outside the site. One of the spent fuel storage pools has capacity to store *451* 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 Seventh 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 *both units* CNA I *and CNA II* 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.

At CNA I in the last *period*, *no waste of the radioactive liquid stream was necessary to treat by concentration, and* 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 CNA II since start of operation it was not necessary to treat the radioactive liquid stream in the concentration system of the Plant (KPF) due its low activity level.

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 NPPs, 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.

No waste generated in the Nuclear Power Plants is taken to the AGE in Ezeiza but is stored temporarily in the Nuclear Power Plants themselves.

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 carried out in support the CNE return to service after the refurbishment activities, shows the following:

- *Regarding the generation of waste, there were no emissions of gaseous or liquid waste higher than normal emissions.*
- *The main radioactive waste corresponded to solid waste.*
- *Regarding the characterization of solid radioactive waste, direct gamma spectrometry was performed on the packages and on samples to determine the activity values and through estimates, scale factors and direct and indirect calculations according to each particular case.*
- *The largest volume of radioactive waste processed during 2019 and part of 2020 corresponded to remnants from the refurbishment of the plant.*
- *The Radioactive Waste Storage Deposits were at the limit of their capacity, so during 2021 the construction of a new deposit for waste, compactable and non-compactable, began.*

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 2019 to December 2021, 259.9 m^3 of radioactive waste were generated.

3.19.9.5. RESEARCH AND DEVELOPMENTMENT ACTIVITIES

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 modelling in sedimentary environments and from the unsaturated zone.
- Hydrogeochemistry characterization studies: hydrogeological, groundwater and geomorphological in sedimentary environments, whose knowledge 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 non-destructive 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 aluminium-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, 2017, *and 2020)*. 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 by 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 (and under recent development) measures are presented in a summary form.

4.1. STRENGTHENING SAFETY CULTURE AT ARN

The Strategic Plan for the years 2021-2025, issued by the ARN's Board of Directors establishes the promotion of a permanent Safety Culture as one of the a strategic lines of the ARN, set up the general objective for development and implementation of a Safety Culture Program integrated in the management system of the organization, and defining the actions to be taken in order to improve and strengthen the current ARN capacities in this regard.

As it is stated in the SARIS Report for the IRRS mission, ARN recognizes that explicit assessments of safety culture are currently pending.

The ARN is planning to undertake self-reflection and self-assessment activities in order to determine the strengths and opportunities for improvements in fulfilling their responsibilities and to enhance the own culture as a regulator with the purpose to positively impact on the licensee's safety culture.

The ARN is aware that performing regular self-reflections, self-assessments, and external reviews, as well as adopting a learning attitude, are key activities to identify areas of improvement in all regulatory activities and support continuous improvement.

4.2. CARRY OUT THE ARN'S ACTION PLAN

In accordance with the IRRS guidelines, ARN developed an Action Plan as part of the measures to address the issues arisen from the self-assessment activities, in preparation to receive the review mission.

The ARN will amend this Action Plan following the IRRS mission to incorporate the recommendations and suggestions contained in the Mission Report and will develop a final Action Plan.

The ARN has strong commitment to identify and implement the corrective actions and activities with the purpose to address the mission's findings in a timely manner.

4.3. CREATION OF A COOPERATION AGREEMENT FOR THE DEVELOPMENT OF THE INDEPENDENT OVERSIGHT FUNCTION ACTIVITIES IN THE LATIN AMERICA REGION

In 2017, a cooperation agreement was created for the development of the Independent Oversight function in Latin America, called Lat-INOS. The member countries are: Argentina, Brazil and Mexico with the participation of Nucleoeléctrica Argentina S.A., Electrobras Termonuclear S.A. - Electronuclear and Comisión Federal de Electricidad (CFE) representing each of the countries mentioned above (respectively). The cooperation agreement has obtained collaboration from the IAEA (RLA 9089).

The main objective of this practice is to establish an external and independent control mechanism focused on safety, reliability, and emergency preparedness, and in non-exclusive way, in the functional, trans-functional, organizational and corporate areas of the company.

The form of implementation is through "peer review missions" carried out through reciprocal visits between the parties, with at least one mission per year.

The review standards are given by the industry best practices, the International Atomic Energy Agency (IAEA) Safety Standards (SS) and the World Association of Nuclear Operators Performance Objectives &

Criteria (PO&C) document (WANO) and the guidelines for independent oversight GL 2018-01 (developed along with the IAEA).

In relation to the peer reviews, up to date, numerous missions have been carried out in Brazil (3), in Argentina (3) and Mexico (2). The addressed topics have included aspects of Leadership, Equipment Reliability, Performance Improvement, Effectiveness in the Resolution of previously identified Improvement Areas, Effectiveness, and Independent Oversight function.

In the case of NA-SA, since 2021, the agreement was used to implement the Nuclear Safety Review Board (NSRB), a review conducted by senior staff that provides NA-SA's Board of Directors with a critical and external view on:

- *The organization's focus on nuclear safety and safety culture.*
- *Stations and corporate organization performance.*
- *Independent Oversight performance.*

ANNEXES

ANNEX I

DRAFT COUNTRY REVIEW REPORT FOR ARGENTINA

EIGHT REVIEW MEETING ON THE CONVENTION ON NUCLEAR SAFETY

RAPORTEUR REPORT

EXECUTIVE SUMMARY

Argentina has 3 nuclear power reactor units. 3 are in operation, 1 is being planned, 1 is under construction. These are the types of nuclear power reactors (3 PHWRs, 1 PWRs, 1 HPR).

X out of 5 Challenges from the 7th Review Meeting have been closed.

The Country Group highlights the following measures to improve safety in Argentina's national nuclear programme:

- A dry storage facility for spent fuel elements is being constructed as an extension of the spent fuel bay building. A sector of the silo will also serve to store irradiated reactor internal components.
- 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 operators to fulfil the fundamental safety functions of shutting down the plant and keeping it in safe state under the event of non-inhabitability of the main control room.
- Actions to optimize fuel management and extending operating time of emergency diesel generators have been implemented, including the development of appropriate instructions.
- The firefighting system of the construction site was identified as an alternative water reservoir to provide water to the spent fuel pools and SGs during a severe accident situation. Plant modifications were implemented to allow this connection.
- Improvements to the physical protection system aimed at strengthening the protection surveillance measures at Embalse in relation to the life extension project.
- Construction of a reservoir with an assured 10,000 tons of water in the discharge channel in case of loss of the lake.
- Spent fuel storage pool level and temperature can be acquired outside the service and reactor building, without requirement of an electrical source.
- An EQ program based on that of Hydro Quebec was adapted to the CNE by considering degree of applicability and the suitability of existing plant documents.
- The steam generators and digital control computers at CNE were replaced as part of life extension work.
- In preparation for phase B of the life extension of CNA I engineering studies are being developed and design upgrades are being analysed based on the PSR's Global Assessment methodology approved in phase A of the life extension plan. It is expected that the Global Assessment and derived Conceptual Implementation Plan will be finished and submitted to ARN by March 2020.

The Country Group highlights the following results of international peer review missions of Argentina:

- The NPP licensee is a member of the World Association of Nuclear Operators (WANO) and during the reporting period all operating NPPs were peer reviewed by WANO with focus on safety and reliability. Some WANO peer reviews are planned for the next reporting period.
- Argentina will host an IRRS mission during the next reporting period.

The Country Group identified the following Challenges for Argentina:

- TBD

In addition the country group identified [#] Suggestions, [#] Area of Good Performance and [#] Good Practices.

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 8th CNS Review Meeting, and therefore complies with Article 24.1.
- Held a national presentation and answered questions, and therefore complies with Article 20.3.

1. BASIC INFORMATION ON ARGENTINA'S NUCLEAR PROGRAMME

Argentina has 3 nuclear power reactor units in operation. These are the types of nuclear power reactors (3 PHWRs). A low power prototype reactor, Central Argentina de Elementos Modulares (CAREM), is under construction at Atucha site.

A Memorandum of Understanding was signed in November 2018, between ARN and NA-SA, oriented to the construction of a fourth NPP. The fourth plant would be an 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.

2. FOLLOW-UP FROM PREVIOUS CNS REVIEW MEETING

2.1. Challenges

Argentina provided the following updates on Challenges identified during the 7th CNS Review Meeting:

Challenge 1: The regulatory authority to prepare and host the IRRS mission in 2018.

Argentina addressed this Challenge by identifying a core group to coordinate the process, defined working methods, working groups and responsibilities. A self-assessment process was also developed utilizing the IAEA SARIS methodology. The mission is scheduled to be hosted May 4th, 2020.

Follow Up Status: Open, Closed,…

Challenge 2: The country to prepare and conduct relevant activities related to CNA I life extension with respect to the SALTO mission.

Argentina addressed this Challenge by hosting two Pre-SALTO missions during the reporting period. The mission concluded that the plant made important progress towards implementing a systematic ageing management review and to prepare the plant for safe long term operation (LTO). This included establishing the regulatory requirements for the definition of LTO. The SALTO mission is scheduled for 2020.

Follow Up Status: Open, Closed, …

Challenge 3: Resolution of issues with CNA I and II RPV in-vessel retention and external cooling arising from FORO stress tests.

Argentina addressed this Challenge by conducting analyses in relation to this strategy and its effectiveness. 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.

Follow Up Status: Open, Closed, …

Challenge 4: The regulatory authority to conduct licensing activities on CAREM 25 small modular prototype reactor under construction following principle 1 of the VDNS.

Argentina addressed this Challenge by continuing to perform assessments, inspections and audits following a safety graded approach. Regulatory activities featured an enlarged scope for the purpose of analysing multiple failure events, with very low probability of occurrence. The design features of CAREM 25 have an improved implementation of the Defence in Depth concept and therefore can be considered an example of how principle 1 of the VDNS can be implemented in future projects.

Follow Up Status: Open, Closed, …

Challenge 5: External emergency control center located far from Embalse NPP.

Argentina addressed this Challenge by working to establish an agreement between NA-SA and the Almafuerte Firefighters Station located more than 15km from the NPP. The temporary municipal emergency control center will be equipped following the instrumentation and communication system to fulfil all of the requirements up to the construction of the definitive facility.

Follow Up Status: Open, Closed, …

2.2. Suggestions

No suggestions were made for Argentina, therefore this section does not apply.

3. MEASURES TO IMPROVE SAFETY

3.1. Changes to the regulatory framework and the national nuclear programme

Since the last Review Meeting, the Country Group took note of the following changes to the regulatory framework and the national nuclear programme.

- Standard AR 10.12.1. "Radioactive Waste Management" was revised to update concept and requirements included in international regulations and, as far as was relevant, the recommendations of the IAEA were taken into account.
- 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.
- ARN conducted a review of their regulatory framework and developed an action plan to be completed in the 2017-2022 period, which includes the creation of 18 new standards and 7 new guides as well as the revision of 11 standards and 1 guide.

3.2. Safety improvements for existing nuclear power plants

The Country Group took note of the following implemented and planned safety measures for existing nuclear power plants in Argentina:

- A dry storage facility for spent fuel elements is being constructed as an extension of the spent fuel bay building. A sector of the silo will also serve to store irradiated reactor internal components.
- 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 operators to fulfil the fundamental safety functions of shutting down the plant and keeping it in safe state under the event of non-inhabitability of the main control room.
- Actions to optimize fuel management and extending operating time of emergency diesel generators have been implemented, including the development of appropriate instructions.
- The firefighting system of the construction site was identified as an alternative water reservoir to provide water to the spent fuel pools and SGs during a severe accident situation. Plant modifications were implemented to allow this connection.
- Improvements to the physical protection system aimed at strengthening the protection surveillance measures at Embalse in relation to the life extension project.
- Construction of a reservoir with an assured 10,000 tons of water in the discharge channel in case of loss of the lake.
- Spent fuel storage pool level and temperature can be acquired outside the service and reactor building, without requirement of an electrical source.
- An EQ program based on that of Hydro Quebec was adapted to the CNE by considering degree of applicability and the suitability of existing plant documents.
- The steam generators and digital control computers at CNE were replaced as part of life extension work.

- In preparation for phase B of the life extension of CNA I engineering studies are being developed and design upgrades are being analysed based on the PSR's Global Assessment methodology approved in phase A of the life extension plan. It is expected that the Global Assessment and derived Conceptual Implementation Plan will be finished and submitted to ARN by March 2020.

3.3. Response to international peer review missions

The Country Group took note of the following implemented or planned measures in response to the results of international peer review missions:

- The NPP licensee is a member of the World Association of Nuclear Operators (WANO) and during the reporting period all operating NPPs were peer reviewed by WANO with focus on safety and reliability. Some WANO peer reviews are planned for the next reporting period.
- Argentina will host an IRRS mission during the next reporting period.

4. 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 to the 7th and the subsequent Review Meetings.

Argentina reports the following safety improvement to the existing nuclear power plants:

- ARN requires the submission of a PSR prior to relicensing. The PSR identifies safety upgrade and improvement areas to be completed during the next licensing period.

Argentina reports the following enhancements to its regulatory framework for the design, siting and construction of new nuclear power plants:

- A memorandum of understanding (MOU) was signed between NA-SA and ARN. This MOU details the regulatory requirements and expectations in terms of the licensing process and safety levels that must be fulfilled by the design of the proposed plant and demonstrated through the Safety Analysis to be further submitted to ARN for the proposed fourth nuclear power plant.
- ARN is in an on-going process of harmonizing the Argentinean Regulatory Standards and the IAEA Safety Standards, although they are already consistent in general terms.

Argentina reports the following planned activities related to the principles of the VDNS:

- The CAREM 25 design features have an enhanced implementation of the Defense 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.
- Operating experience from feedback programs, internal and external events and research findings are taken into consideration during operation and commissioning stages. It is required that the results of the implementation of these programs are submitted to the regulatory body.

The Country Group made the following observations:

- TBD

5. RESULTS OF THE REVIEW

5.1. General Quality of the National Report

With regards to the general quality of the National Report and transparency issues, the members of the Country Group made the following observations:

- The Report is qualified to be comprehensive and reader friendly.
- The report was of good quality.

With regards to the compliance with the requirements of the CNS and its Guidelines, the members of the Country Group made the following observations:

- The Report was submitted on time for the deadline of August 15th, 2019.
- The content and structure of Argentina's National Report complies with the CNS guidance.
- The directions of the Summary Report of 7th Review Meeting were taken into consideration.
- The directions given by the President of the 8th Review Meeting were followed.

5.2. Participation in the Review Process

With regards to Argentina's participation in the Review process, the members of the Country Group made the following observations. Argentina

- posted questions to Contracting Parties.
- delivered answers to the questions of Contracting Parties on time.
- delivered its national presentation.

5.3. Challenges

The Country Group identified the following Challenge(s) for Argentina.

- Challenge 1: TBD

5.4. Suggestions

The Country Group identified the following Suggestion for Argentina.

- Suggestion 1: Argentina should consider a benchmarking of the Ageing Management plans, for example with those developed in other CANDU countries.

5.5. 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 Area of Good Performance.

The Country Group identified the following Good Practices:

- Good practice 1: TBD

The following Area of Good Performance of Argentina were commended by the Country Group:

- Area of Good Performance 1: Providing technicians with in-depth radiation protection training (8 weeks, 7 hours a day) in addition to on-the-job training.

6. FULFILMENT OF CNS REVIEW REQUIREMENTS

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 8th CNS Review Meeting, and therefore complies with Article 24.1.
- Held a national presentation and answered questions, and therefore complies with Article 20.3.

ANNEX II

ANSWER TO QUESTIONS OR COMMENTS NATIONAL NUCLEAR SAFETY REPORT – 2019

No. 1
COUNTRY: UNITED KINGDOM
CNS-REF.-ART.: Article 6
PAGE OF REPORT:
CHAPTER OF NAT. REPORT: 3.6.5.2

The Eighth report lists amongst the activities to be carried out in preparation for the long term operation of CNA I that demonstration of "Leak before Break behaviour" will be undertaken.

Please describe what this demonstration will entail, and how the design would be demonstrably resilient to the consequences of gross failure (double-ended guillotine break).

The CNA I LTO programme involves the demonstration of LBB on the main piping of the reactor. This demonstration is based on the analysis previously performed for the CNA II commissioning. It entails the following tasks:

- Scope and screening of the piping to be analysed. Identified systems: main cooling and moderator circuits, pressurizer surgeline and sprayline.
- Collecting and verification of the documents related to materials for the LBB.
- LBB Fractomecanic assessment for each identified system.
- Additional measurements for leak detecting. Tritium detectors.

Drawing from the analysis conclusions, CNA I foresees that additional analyses and/or actions could be needed (i.e., increase the ISI frequency on some welds, reduce conservatisms on the stress calculations, etc.).

No. 2
COUNTRY: UNITED KINGDOM
CNS-REF.-ART.: Article 10
PAGE OF REPORT:
CHAPTER OF NAT. REPORT: 3.10.2.2

The Eighth report outlines activities of the Licensee on safety culture (SC) such as the Programme of Consolidation of Safety Culture (PRACS) and the role played by ARN in evaluating safety culture during inspections. The report, however, does not appear to include the outcomes, trends, any challenges or specific actions arising from safety culture evaluation by either the Licensee's or ARN activities.

Please outline the key findings and actions arising from the safety culture evaluations.

As a result of the last safety culture self-assessment carried out for the Licensee at corporate level, three areas for improvement were identified and certain global actions were proposed as follows:

AFI 1: Identification of problems and troubleshooting

"[…] the effectiveness of corrective actions is not always measured and in some cases there are expired corrective actions. On the other hand, not all indicators have defined objectives or reflect the process they monitor."

Proposed actions:

1) Set up methodology for the Effectiveness Analysis of Corrective Actions.
2) Define corporate expectation for rescheduling of due dates.
3) Identify, analyse and reformulate indicators that do not contain clear objectives or do not provide the necessary information for decision making.

AFI 2: Work practices

"… In the execution of field work, a questioning attitude is not always prioritized, the procedures are not always strictly followed, nor are the standards of good work practices maintained in some activities."

Proposed actions:

1) Create a campaign to disseminate the benefits of good practices in the field with the goal of identifying risks.
2) Define a methodology to include either in the Work Packages or in Work Instructions, the Error Prevention Techniques as well as the good practices of the industry.

AFI 3: Coordination of work teams

"[…] the work management process has weaknesses in its programming and in the coordination of the teams. Additionally, differences are found in certain modes of management processes between CNE, CNA I-II and SPC, which hampers its efficiency".

Proposed actions:

1) Proposal to carry out a benchmarking (i.e. to know strengths and weaknesses) with the objective of unifying the planning and programming process among the people in charge of CNE, CNA and SPC.
2) Analyse the possibility of adding Work Management (planning and programming) in the PRACS.

No. 3
COUNTRY: UNITED KINGDOM
CNS-REF.-ART.: Article 14
PAGE OF REPORT:
CHAPTER OF NAT. REPORT: 14.2.1.2.1 CNA

The Eighth report outlines in section 3.14.3.1.3.1 a series of additional analysis, improvements and modifications to be performed on CNA I from the stress tests undertaken. These include, amongst others, the evaluation of the need for a filtered containment system, for the cooling of the exterior of the Reactor Pressure Vessel (RPV), or the installation of the in-core instrumentation for Severe Accident Management. These activities are noted as expected to take place during the Life Extension Project.

Please confirm the expected status of analyses and measures by the time ARN's permissioning decisions are to be made.

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 were analysed and extra efforts had to be made to adjust codes to the Atucha reactors. NA-SA along with ISS, the current developer of RELAP5 / SCDAP, have developed a version of the code that is representative of the expected phenomenology in Atucha reactors (RELAP5 / SCDAP Mod 3.6). In the past years, preliminary results had been obtained with RELAP5 / SCDAP. These calculations were followed by more complex analysis with ANSYS / CFD code, performed for CNA II NPP. The results of these analyses showed that the countermeasure is not successful in a scenario of LOCA in the moderator circuit with failure of safety injection system or in a SBO scenario. Based on these results, it was decided to rule out this countermeasure for Unit II and Unit I. It should be noted that the results for Unit II are extrapolable to Unit I in this case. As it was mentioned, a parallel course of analysis has been started, to assess possible stabilization of molten material inside sump; avoid an early containment breach; and to decrease consequences in public as far as reasonably achievable. This task is being performed as part of phase B of CNA I LTO. This project is part of the Conceptual Improvement Plan that will be presented for consideration to ARN in March 2020 (see section 3.6.5.2).

Regarding the venting filtered containment system it is also planned to be implemented as part of phase B of CNA I LTO.

No. 4
COUNTRY: UNITED KINGDOM
CNS-REF.-ART.: Article 17
PAGE OF REPORT:
CHAPTER OF NAT. REPORT: 3.17.2.3.2.1.2

The Eighth report states in section 3.17.2.3.2.1.2 that the minimum and maximum water levels at the Atucha I and II have been reassessed and that the extreme river level rise of Río de la Plata considered was a 1 in 1,000 year event. For natural external hazards characterised by frequency of exceedance hazard curves, it is generally expected that a design basis event should be derived conservatively to take account of data and model uncertainties. An event with an annual probability of exceedance of 10^{-4} is frequently used for non-discrete hazards.

Given the above, please justify the approach taken including the frequencies selected. How has the impact of climate change been factored in?

NA-SA requested INA (National Water Institute) to develop a re-assessment of Paraná River Extreme Levels. The study took into account historical records and previous studies to simulate different scenes adding catastrophes (e.g. Itaipú and Yaciretá dams burst) and extreme meteorological events (storms and others).

One of the main references was “Analysis of Extreme Hydrometric Dimensions at the Nuclear Power Plant Site - Atucha II Nuclear Power Plant”, which applied the Gumbel distribution (Generalized Extreme Value distribution Type-I) to model the maximum level distribution of Paraná River. It was found that Log-Pearson Type III distribution had a better fit for minimum value. For these reasons it was the selected distribution for downspout.

On the other hand, the report from INA took into account the study “Risk analysis of flood duration in the coastal areas of the Río de la Plata considering climate change”. This study models the influence of the Meteorological event called sudestada (strong winds from southeast) over the La Plata River levels and Paraná River regime.The implicantions of this model can be extrapolated to recurrence periods of 1.000 years.

The impact of climate change was also considered in the aforementionted study, where possible scenarios for 2030 and 2070 are modelled, predicting a major influence of south-east winds (higher levels and longer duration). INA suggests that CNA should make a follow-up of the Río de La Plata basin evolution. It merits noting that CNA already makes a continuous tracing and record of Paraná River levels. These data can be checked out with the data released by governmental services, like National Naval Prefecture.

No. 5
COUNTRY: ICELAND
CNS-REF.-ART.: Article 16
PAGE OF REPORT:
CHAPTER OF NAT. REPORT: Article16

Does Argentina intend to complete a self-assessment on its emergency preparedness and response arrangements with regard to nuclear and radiological emergencies based on the EPRIMS tool and to share information on the results?

Argentina has already made its self-assessment with EPRIMS tool. The last update was at the end of 2019.

No. 6
COUNTRY: LUXEMBOURG
CNS-REF.-ART.: Article 16
PAGE OF REPORT: 128
CHAPTER OF NAT. REPORT: 3.16.8

On page 66 the report highlights the interaction with the Uruguayan and Brazilian Regulatory Bodies in their character of neighboring country, with regard to their participation in the practical emergency plan exercises at CNA as a possible good practice. On page 128, it is stated that

representatives from those two states took part as observers in some of the exercises. In relation to those statements, we would like to ask two questions:

1) **What is the main goal to invite those representatives as observers?**
2) **Given that the Atucha NPP is in a relative proximity to Uruguay, are there any other bilateral mechanisms in place to exchange information or to collaborate bilaterally in case of a nuclear emergency?**

1) The invitation to external observers is done with the purpose to offer them an opportunity to provide feddback which can inform future exercises or in real situations. By the same token, those feedback comments can be useful in exercises carried out in their respective countries as well, putting into practice tasks developed in our scenarios.
2) Yes. Regarding the EPR, there are mechanisms to share information with Uruguay. Moreover, cooperation is also possible through the IAEA exchange mechanisms in compliance with the Prompt Notification and Assistance Conventions.

No. 7
COUNTRY: PAKISTAN
CNS-REF.-ART.: General
PAGE OF REPORT:
CHAPTER OF NAT. REPORT: 2.4

Reference section 2.4 of the report, the licensing of CAREM 25 small modular prototype reactor was highlighted as challenge for Argentina. Argentina may like to elaborate this challenge with specific examples.

Regarding the licensing of CAREM 25 prototype reactor, the most relevant/salient challenges and issues faced by ARN are described as follows:

- The methodology for the Safety Classification of Structures, Systems and Components has a systematic methodology and clear involvement in Defense in Depth. It allows to establish engineering requirements to ensure an adequate functional capacity, robustness and reliability of the relevant SSCs.
- Safety demonstration through modeling and computer simulation for SSC within a consolidated experience. V&V of TH and neutronics codes.
- Safety demonstration through tests in an installation with adequate similarity, used for parametric studies of the dynamic response to perturbations, changing steam dome volume, hydraulic resistance, P and T.
- Construction inspections and regulatory audits.

No. 8
COUNTRY: PAKISTAN
CNS-REF.-ART.: General
PAGE OF REPORT:
CHAPTER OF NAT. REPORT: 2.4

Reference section 2.4 of the report, it is mentioned that 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. Argentina may like to share detail of areas that need special attention during licensing of passive safety systems.

As mentioned in section 2.4, one of the areas with special focus is the validation and verification of computer codes used for the Responsible Entity for the demonstration of Safety.

ARN does not have specific standards for the system codes used for safety analysis/ demonstration (TH, neutronics, fuel behavior, etc.).

Most commonly used codes have been already validated and verified within the respective correlations range (RELAP5, TRACE5, CATHENA, CATHARE, MARS, ATHLET, and others).

However, due to special design features of CAREM reactor (e.g. the Passive Safety Systems), ARN required a comprehensive analysis of each condition and physical phenomena that could occur during PIEs transients/accidents and checking codes capability for capturing and representing them.

This procedure holds mainly for thermal-hydraulics and neutronics codes (cell and core level). Some specifics codes have been developed for engineering/design, e.g. steady-state conditions, DNB, instabilities maps.

ARN will provide/provides guidance on how to develop models/codes, its documentations and the quality assurance process required (based on CNSC RG G-149).

No. 9
COUNTRY: CHINA
CNS-REF.-ART.: General
PAGE OF REPORT: 13
CHAPTER OF NAT. REPORT: 2.5

How to decide the site selection and construction standard of CNE External Emergency Control Centre, including the safety assessment of distance from the Embalse NPP and the seismic level?

For the External Emergency Center, a construction report was requested from the Nuclear Power Plant, ARN analyzed this report. Regarding its location, building will take place outside the UPZ (10 km). The design and construction characteristics have been evaluated as well as specific safety conditions have been considered.

No. 10
COUNTRY: CHINA
CNS-REF.-ART.: General
PAGE OF REPORT: 15
CHAPTER OF NAT. REPORT: 2.7

CNA 1 was reached the end of design life in April 2018 and maintain the current licensing basis as defined by the FSAR and PSR performed in 2014, how long does this stage last?

The current Operating Licence was amended in April 2018 with the purpose to authorize the operation beyond the design lifetime. The validity period of this Licence is five full power years (5 FPY) or until September 29th 2024, whatever occurs first. This last date corresponds to the expiration date of the PSR performed in 2014 (10 years cycle).

No. 11
COUNTRY: SWEDEN
CNS-REF.-ART.: Article 7
PAGE OF REPORT: 32-33
CHAPTER OF NAT. REPORT: 3.7.2

What process is used to prepare AR Standards? Are proposals for new AR Standards submitted to relevant licensees and expert organizations before they are adopted?

ARN follows an established process for the elaboration, review and revision of standards and guides in the framework of ARN management system.

Regulatory Standards Division, which reports directly to ARN Board of Directors, coordinates the activities of the process for elaboration, review and revision of standards and guides.

The process for any standards includes the following steps:

- The evaluation of the need to develop / revise standards and guides, that is performed by technical or management areas based on the regulatory experience; operator experience, international recommendations and the requirements of the conventions signed by Argentina,
- The participation of ARN senior staff of technical areas to elaborate the initial drafts and revisions process,
- The call for the interested parties involvement (not only licensees but any human or legal persons who felt affect by the standard) before standard approval and
- The access to published Standards and Guides on the ARN website.

When ARN plans to issue or modify a standard applicable to relevant licensed facilities, in addition to the previous steps, a Prior Consultation Procedure has to be applied. According to this procedure, the main licensees of said facilities are invited to join a committee to evaluate the proposed standard, and then all licensees are able to give their opinion on the project of standard before its approval.

No. 12
COUNTRY: SWEDEN
CNS-REF.-ART.: Article 7
PAGE OF REPORT: 32-33
CHAPTER OF NAT. REPORT: 3.7.2

Does your national framework address questions about safety-security interface in nuclear facilities? If so, how is this done?

National Framework includes requirements related to nuclear and radiological safety, safeguards and security.

All the standards of the ARN establish that its compliance does not relieve from compliance with other norms and requirements established by both the ARN and other competent authorities.

The standard AR 10.13.1. “Basic standard for the physical protection of nuclear materials and installations” establishes that no physical protection measure must be at expense of radiological and nuclear safety. Also, it is required that technical procedures and activities related to plant operation must be taken into account in the design of the physical protection system.

The standard AR 10.13.2. “Security of radioactive sources” establishes that no any security measure must be at the expense of radiological and nuclear safety.

It is foreseen that new revisions of the standards related to radiological and nuclear safety applicable to all life stages of nuclear reactors will explicitly establish requirements regarding interfaces between all regulatory areas, when applicable.

No. 13
COUNTRY: SWEDEN
CNS-REF.-ART.: Article 7
PAGE OF REPORT: 36
CHAPTER OF NAT. REPORT: 3.7.2.3.1

How do ARN handle the difficulties in a performance-based regulatory approach with identification of good outcome measures?

There are several names for the regulatory approach alternative to the Prescriptive Approach: Performance-Based Regulatory Approach; (Safety) Goals Oriented Approach; or simply non-prescriptive. ARN has a sound understanding of both approaches and may be flexible with the names.

There are no particular difficulties in identifying good outcomes in a performance-based regulatory approach. A non-prescriptive approach does not imply relaxing acceptability or success criteria, nor having ad-hoc rules for each assessment (that would produce difficulties the identification of good outcome measures). A non-prescriptive approach implies judging some cases by the safety goals behind the pre-written (pre-scribed) regulatory requirements. Perhaps a definite answer to the question requires ensuring a common understanding of the different regulatory approaches.

ARN understands that a good approach to “good outcome measures” is based on the identification and prioritization of outcomes, being safety the parameter of the prioritization. In this sense, good “outcome measurements” are essentially safety goals oriented. If the outcomes were to be measured against the strict compliance against the wording of an undifferentiated list of requirements (prescriptive), it would be possible to perceive acceptance by a numerical value (let’s say 95% of compliance) even having one or two essential safety requirements not-complied.

No. 14
COUNTRY: SWEDEN
CNS-REF.-ART.: Article 7
PAGE OF REPORT: 37
CHAPTER OF NAT. REPORT: 3.7.3.2.1

How is the Environmental Impact Assessment (EIA) addressed as part of the licensing process? Which authority is responsible for this being done and what assessments is being done of the EIA in parallel with the nuclear safety assessments?

The Environmental Impact Assessment (EIA) is a requirement included in Argentina provincial legal frames. The Law 11459 (Radicación Industrial) applies in Buenos Aires province, whose regulatory authority is OPDS (Provincial Agency for Sustainable Development), and the Law 10208 (Provincial Environmental Policy) applies in Córdoba province, whose regulatory authority is the Ministry of Water, Environment and Public Services.

These laws determine the mandatory contents of an Environmental Impact Assessment from a non-radiological point of view. Several points of EIA concur with those described in the Final Safety Report, Chapter 2 relative to Site Characteristics e.g.: physiographic characteristics, climate descriptions, socioeconomic condition, connectivity etc. The EIA of any project is sent to the corresponding authority previous to the start of the project, to get the environmental approval.

According to the activity’s particular features, the radiological aspects and impacts control is within the scope of ARN (National Nuclear Regulator) in the frame of Law 24804 (Nuclear Activity Law and its Regulatory Decree 1390/98), and CNEA (National Atomic Energy Commission) as the responsible entity of Law 25018 (Radioactive Waste Management Regime).

No. 15
COUNTRY: SWEDEN
CNS-REF.-ART.: Article 7
PAGE OF REPORT: 37
CHAPTER OF NAT. REPORT: 3.7.3.2.1

Is it part of the applied licensing process to have public (nearby residents, environmental organizations) insight, e.g. through hearings? At what stages?

The licensing process in Argentina doesn’t contemplate a formal participation of the public through public hearings. However, there is a communication procedure through which ARN responds to interested parties' concerns and communicates their actions for example, with regard to licensing of a new NPP. This can be done in any stage of the licensing process upon public’s demand.

No. 16
COUNTRY: SWEDEN
CNS-REF.-ART.: Article 7
PAGE OF REPORT: 39-40
CHAPTER OF NAT. REPORT: 3.7.3.3

Is it included in ANR's planned or special inspections to also follow up the license holders' work with safety culture?

Yes, follow up the license holder’s work with safety culture is part of the inspected topics in a NPP for all type of inspections.

No. 17
COUNTRY: SWEDEN
CNS-REF.-ART.: Article 9
PAGE OF REPORT: 59
CHAPTER OF NAT. REPORT: 3.9.2

"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....."

What measures have been implemented by the licencees to abide by the regulatory standards above?

The Licensee has an integral management system in place which includes all the processes and associated procedures in order to fulfil the mentioned regulatory standards and any other regulatory requirements.

No. 18
COUNTRY: SWEDEN
CNS-REF.-ART.: Article 9
PAGE OF REPORT: 61
CHAPTER OF NAT. REPORT: 3.9.3.5

"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."

Is not 7000 visitors in one year a quite high number? Has the increased security requirements, put in place after 9/11, affected how plant tours are conducted?

Of the total visits received at the Atucha Nuclear Complex and Embalse NPP, about 90% of them (89% in the CNE and 97% in the CNA) correspond to educational institutions, mainly secondary schools.

Only 1% of visitors in Atucha and 8% Embalse are individuals who come to visit the plants as part of our tour program, which implies a previous registration that involves entering a waiting list for the tour.

All of our visitors, students, individuals, authorities and officials, must go through the physical security instances of the plants before entering, in addition to having a rigorous control of data prior to the visit.

Before taking the tour through the plants, a presentation is held where the public interacts and, if necessary, any type of attitude that does not fit the normal parameters of a visitor are observed and reported.

We consider it a great pillar of our approach to society opening the doors of the plants and showing the community our work and it is in our interest to strengthen this program to continue strengthening our links with the population near the plants.

No. 19
COUNTRY: SWEDEN
CNS-REF.-ART.: Article 9
PAGE OF REPORT: 60
CHAPTER OF NAT. REPORT: 3.9.3.1

It is written that the intention of communication with the public is to install a positive perception of the nuclear power generation.

Is this really the main and only goal of the communication? What are the other main goals with communication to the public?

"Installing a positive perception about the generation of nuclear energy in favor of the operation and continuity of nuclear power plants operated by Nucleoeléctrica Argentina is one of the objectives of communicating with the public but not the only one.

Nucleoeléctrica Argentina also aims to strengthen its image as a utility that produces baseload, safe and clean energy; promote the development of nuclear technology as a solution to the effects of climate change; carry out the communication of the projects that the company develops; inform and familiarize the various stakeholders about the safe and responsible operation of nuclear power plants; maintain an open and transparent communication with interested parties; and provide communication support in crisis and emergency situations, among other objectives."

No. 20
COUNTRY: SWEDEN
CNS-REF.-ART.: Article 10
PAGE OF REPORT: 66
CHAPTER OF NAT. REPORT: 3.10.2.2

"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."**

In practice, what activities are performed to achieve the communication and dialogue described above?

In practice, the communication and dialogue as described in section 3.10.2.2. are reached by on the job coaching of personnel. Coaching is considered one of the most powerful attributes for effecting change. Coaching helps influence “the people variable” in the change process towards a professional regulator-operator relationship.

No. 21
COUNTRY: SWEDEN
CNS-REF.-ART.: Article 10
PAGE OF REPORT: 66
CHAPTER OF NAT. REPORT: 3.10.2.2

The Licensee 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 stations.

What kind of improvements have you found by using PRACS?

"Among the improvements found using PRACS, it can be mentioned: process and procedures unification between both NA-SA power plants sites (eg, error prevention techniques, corrective actions, operational decision making, company emergency plan, self-assessments, indicators); development of plans for joint communication between the 2 NA-SA power plants sites (eg industrial safety topics), cross-sectional application of WANO guides throughout the organization (eg Human Performance Program Assessment Model).

All the above-mentioned improvements produce tangible products, but in turn the PRACS, through regular meetings between the different experts on different topics, generates an intangible but very important added value, linked to the promotion of integration and communication between the different company areas, thus avoiding isolated areas of knowledge."

No. 22
COUNTRY: SWEDEN
CNS-REF.-ART.: Article 12
PAGE OF REPORT: 74
CHAPTER OF NAT. REPORT: 3.12.1

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.

Which requirements regulate 'the field training'?

The reinforcement of expectations and practices of field work complements the requirements regulated by the initial and continuous training programs. In this way, they are transformed into brief spaces of exchange between the leader and their team, in which the necessary alertness is maintained and increased so that the behaviours are a solid barrier in the different activities. Additionally, the findings that arise in the field and can be improved with training are notified to the training department so that it has field information when developing its continuous training programs.

No. 23
COUNTRY: SWEDEN
CNS-REF.-ART.: Article 12
PAGE OF REPORT: 74
CHAPTER OF NAT. REPORT: 3.12.1

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.

How is intervention prohibited "during the period immediately after" (grace time) practically regulated?

The regulatory authority requires that the protection system be automatically initiated in the case of a postulated event. The design of the protection system shall automate various safety actions to actuate safety systems so that operator action is not necessary within a justified period of time from the onset of anticipated operational occurrences or accident conditions. This means that operator intervention is not required, because the system is capable of executing all safety measures needed. On the other hand, the prohibition of operator intervention is regarding any action that would result in an inhibition or interruption of any safety measure executing during the grace period. This is a requirement demanded by the regulatory body to the system’s design, and is assessed and approved at the design phase of the project. In this way, the regulatory body ensures that there is no human intervention aiming to stop or prevent any safety measure taken by the system as a result of the system not allowing it by design.

No. 24
COUNTRY: SWEDEN
CNS-REF.-ART.: Article 12
PAGE OF REPORT: 75
CHAPTER OF NAT. REPORT: 3.12.1.1

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.

How was the improved, over-all plant safety measured?

The improvement in human behavior is monitored by the presence of leaders in the field and by the analysis of trends in internal events on CNA, with special emphasis on those internal events with causes associated with behavior. Monitoring these trends and the general actions for common findings of different internal events is the way to monitor and improve safety in work practices.

No. 25
COUNTRY: SWEDEN
CNS-REF.-ART.: Article 12
PAGE OF REPORT: 75
CHAPTER OF NAT. REPORT: 3.12.1.1

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

COMMENT: Good examples of programme points to evaluate incidence of human factors, especially points 1 and 3.

Argentina appreciates the comment from Sweden.

No. 26
COUNTRY: SWEDEN
CNS-REF.-ART.: Article 12
PAGE OF REPORT: 76
CHAPTER OF NAT. REPORT: 3.12.1.1

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.

How often does training and retraining of all personnel occur?

Training and retraining in human performance is defined in the initial and continuous training programs for each position. Additionally, there are 52 weekly human performance reinforcements which are disseminated by each group leader. There, different topics are refreshed, previously communicated expectations are reinforced, results of previous performance are spread and expectations that are pursued in current performance are reiterated.

No. 27
COUNTRY: SWEDEN
CNS-REF.-ART.: Article 12
PAGE OF REPORT: 78
CHAPTER OF NAT. REPORT: 3.12.1.3.1

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.

The 12 hour retraining - is it a "one time action" or is it periodically repeated?

The “Back to Basics" Workshop was a one time Human Performance program for all plant personnel.

No. 28
COUNTRY: SWEDEN
CNS-REF.-ART.: Article 12
PAGE OF REPORT: 79
CHAPTER OF NAT. REPORT: 3.12.2

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.

The NPP’s Primary Responsible determines the responsibility of persons for errors and, if applicable, applies pertinent sanctions. Please explain how applying this policy leads to prevention of undesirable events? Could such application of sanctions not lead to that errors are not being reported (hiding of the facts)?

Argentina considers that prevention of undesirable events is not achieved through a sanctions regime. Instead, it relies on the activities mentioned in the whole Article 12.

Please understand those paragraphs in the framework of the specific actions at each nuclear power plant that are described in sections 3.12.1.1.3.12.1.2 and 3.12.1.3.

The application of sanctions is the last resort used and has to be graduated with the severity of the infraction.

No. 29
COUNTRY: SWEDEN
CNS-REF.-ART.: Article 13
PAGE OF REPORT: 82
CHAPTER OF NAT. REPORT: 3.13.2

Periodically, the QA Management issues reports showing the audits’ results. It is also stated: "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".

Are there requirements on the periodicity of QA activities?
Is it really possible to train the staff every time a new revision is performed?

1) The audits are annually planned on the basis of a series of factors, such as: related issues found in previous audits, problems performing tasks, importance of the processes with respect to safety, new projects, result of process indicators, new requirements, etc.
2) Following a graded approach to safety, the training requirements after a revision of a certain procedure is performed are met. According to the established procedures, the sector responsible for the modification or generation of a procedure determines, together with the training sector, which participants need to receive training as well as the modality to be used, that is, in a classroom or through the intranet. When the latter is performed, the participants included in the training receive an e-mail communication with a link to access the online e-learning platform.

No. 30
COUNTRY: SWEDEN
CNS-REF.-ART.: Article 14
PAGE OF REPORT: 94-95
CHAPTER OF NAT. REPORT: 3.14.3.1.3

To our understanding, plant damage state characterization is still underway, i.e. the SAMG are still based on five specific scenarios (small LOCA etc.). Is this correct?

One of the main principles in IAEA Safety Guide NS-G-2.15 say that "Accident management guidance should be set out in such a way that it is not necessary for the responsible staff to identify the accident sequence or to follow some pre-analysed accident in order to be able to execute the accident management guidance correctly". When are the guidelines based on plant damage state planned to be implemented?

1) The “five specific scenarios” mentioned were just an initial constraint within the regulatory requirement of developing a Severe Accident Management Program. That philosophy is superseded these days.

   In the present, SAMG are based primarily on both deterministic and probabilistic assessments, and where a lack of them is unveiled, engineering judgment is applied. By these means, challenges to safety functions or barriers are identified and plant vulnerabilities are determined. Viable SAMG are then developed taking into consideration those insights.

   <u>CNA II</u>

   An accident progression model is developed, and continuously upgraded, with MELCOR 1.8.6 code. Also, a RELAP5/SCDAP Mod 3.6 model is used for assessing the in-vessel stage phenomenology and for SAMG verification.

   A full PSA (L1, L2 & L3) is developed and constitute the basis for determining sequences for which feasible SAMG might be considered.

   <u>CNA I</u>

   CNAI counts with a preliminary accident progression model coded in MELCOR. It is complemented with a RELAP5/SCDAP Mod 3.6 to provide a better representation of the thermal-hydraulic behavior of the plant and to address in-vessel phenomenology. Also, evaluation of SAM strategies is performed.

   PSA L2 for CNA I was not developed yet, but it is expected to be elaborated within the life extension project. In the meantime, experience from CNA II PSA L2 was transferred to CNA I in order to develop SAMG. Also, a multi-compartment model of CNA I containment with GOTHIC 8.1 is being tested to assess SAM strategies.

   <u>CNE</u>

   SAMG for CNE are adapted from the generic guidance provided by COG, which are based on accident progression assessments performed with the MAAP4-CANDU code.

2) In effect, Severe Accident Management Guidelines based on symptom diagnosis are being implemented at the moment in all of our operating NPP.

No. 31
COUNTRY: SWEDEN
CNS-REF.-ART.: Article 14
PAGE OF REPORT: 97
CHAPTER OF NAT. REPORT: 3.14.3.1.3.1: CNA I - Filtered Containment Venting system

"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".

The stabilisation of corium, both in-vessel or ex-vessel, are phenomenon associated with large uncertainties. For the ex-vessel scenario large amount of noncondensible gases can be produced due to corium interaction with basemat concrete, together with intensive steam production. Hence, a containment overpressurisation protection system, as FCVS, is to be considered a reasonably practicable safety improvement of defence in depth level 4. An accident management strategy for DID level 4 typically requires the installation of dedicated

safety systems for the protection of the last barrier and cannot solely be based on analytical efforts due to the complexity and uncertainty of the involved phenomena.

How and when will Argentina act regarding filtered containment venting system, especially in the light of the Vienna Declaration that states that "Reasonably practicable or achievable safety improvements are to be implemented in a timely manner"?

As it was mentioned in the 8th report, CNA I operation is currently under phase ""A"" of long term operation, LTO (see section 3.6.5.2). The filtered containment venting system for CNA I is planned to be implemented as part of the improvements for phase B of CNA I LTO.

No. 32
COUNTRY: SWEDEN
CNS-REF.-ART.: Article 14
PAGE OF REPORT: 94
CHAPTER OF NAT. REPORT: 3.14.3.1.1

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.

What kind of updates, apart from plant modifications, are requested in the five year-update?

What kind of evalutations are required for those analyses that are not updated?

1) For ARN, safety is a dynamic concept and because of that licences are granted for a validity period of 10 years and continued operation is based on the licensee's submission and regulatory approval of a Periodic Safety Review (PSR). Following this approach SAR update has to be performed in order to reflect the upgrade of the licensing basis derived from the improvement measures identified after comparison of the plant against modern standards.

   Currently, ARN is performing an updating of its regulations including AR 3.9.1 and the requirement of updating of the SAR every five years will be modified to ten years in accordance to PSR concept. Regardless the frequency for SAR update, the regulatory expectation is to update the licensing basis aiming at reducing the gap (as far as practicable) of the existing plants in comparison to modern plants.

2) For those analysis that are not updated it is required the evaluation of operational aspects which are relevant to safety. The main aspects are maintenance, surveillance and testing, management of ageing and analysis of operating experience.

No. 33
COUNTRY: SWEDEN
CNS-REF.-ART.: Article 14
PAGE OF REPORT: 99
CHAPTER OF NAT. REPORT: 3.14.3.1.3.2: CNA II - Filtered Containment Venting system

Some filtering system common to both units is being considered.

The Fukushima accident shows that we can have severe accidents affecting all plants at a site. What is the reasoning behind the consideration to have a common system for both plants? Have all the risks connected to such a common system been assessed?

The common containment filtered system for CNA unit I and II was ruled out. Independent filtered systems for unit I and II are planned to be implemented at the same time in order to take the advantages of carrying out similar projects together. Its implementations are foreseen for phase B of CNA I LTO.

No. 34
COUNTRY: SWEDEN
CNS-REF.-ART.: Article 14
PAGE OF REPORT: 107
CHAPTER OF NAT. REPORT: 3.14.3.2.2

For commissioning of CNA II it was required to develop PSA Levels 1 to 3.
Does the PSA study for CNA II cover all plant states?

Currently, the plant states covered in PSA study for CNA II are full power and internal events.

No. 35
COUNTRY: SWEDEN
CNS-REF.-ART.: Article 14
PAGE OF REPORT: 104
CHAPTER OF NAT. REPORT: 3.14.3.1.3.5

In December 2012, CANDU Energy performed a Severe Accident Management Guidance (SAMG) Training and Validation Exercise for the Severe Accident Management Program.
Was the validation for some specific cases or for all plant damage states?
What was the outcome of the exercise? Will the revised guides go through the same validation?

Validation Exercise for the Severe Accident Management Program was done for two selected scenarios because it was impractical to hold a Validation Exercise for every severe accident scenario, or even in order to exercise every SAG, SCG and CA.

The main goals of validation plan were:

- Effectively integrate the SAM function of the TSG into the emergency response organization;
- Provide confirmation that SAM guidelines are viable to implement;
- Provide necessary and sufficient training to all staff involved in response to a severe accident;
- Ensure effective communications between involved participants during a severe accident response.

The selected two specific severe accident scenarios satisfy the following requirements:

- Be among the most likely severe accident scenarios, based on PSA insights;
- Include a transition from EOPs to SAMG;
- Identify the relevant equipment failures and human errors throughout the scenario;
- Involve the use of updated plant status sheets, provided by the controllers when required by any significant event;
- Involve use of EOPs, SACRG-1, SACRG-2, the DFC and SCST, entry into at least one SAG and one SCG, use of one or more related CAs, and SAEG-1 and SAEG-2;
- Require the use of communication methods and plant data management tools that are designed for use during a severe accident response.

To be as realistic as possible, the final validation scenarios has taken into consideration the location of the CICE and TSG, the different groups that are settled in these locations, and the information available in each area. The duties and roles of the different actors of the emergency response organization were detailed (i.e., the control room, field operators, the TSG, the radiation protection group, the site security group, the public information group, the logistic group, etc.).

The outcome of the exercise was satisfactory because the main goals of the validation were verified. As a result of this exercise, a set of minor modifications on the SAM guidelines were generated. The most important modification was changes in the format of some information and tables in order to do it more friendly and easy to use.

The guides were revised after the validation exercise and the mentioned minor changes were included in the revision.

Note:
TSG: Technical Support Group
EOP: Emergency Operating Procedure
SACRG: Severe Accident Control Room Guideline
DFC: Diagnostic Flow Chart
SCST: Severe Challenge Status Tree
SAG: Severe Accident Guideline
SCG: Severe Challenge Guideline
CA: Computational Aids
SAEG: Severe Accident Exit Guideline
CICE: Internal Centre of Emergency Control

No. 36
COUNTRY: SWEDEN
CNS-REF.-ART.: Article 15
PAGE OF REPORT: 114
CHAPTER OF NAT. REPORT: 3.15.2.1 - Table 3.15.1

Activity released from CNA I to the environment as gaseous discharges "CNA I" seems to have a decreasing the I-131 release to the environment from 3.4 x 10^{-3} in 2016 to 8.7 x 10^{-6} TBq in 2018, while "CNA II" release is around 2.7 x 10^{-3} to 7.0 x 10^{-4} Tbq.

Is there a fuel damage policy which affects release of I-131 at the plant? Eg. is there a restriction on allowed tramp uranium on the core or the I-131 release to the environment during operation with damaged fuel? Is there a source term reduction program at the plants and is it connected to fuel failure management?"

1) Within the Plant Policies and Principles Manual there is an operational criterion for activity concentrations when failed fuels are detected in the core.
2) They are defined in the operating specifications of the plant with failed fuel elements.
3) ALARA Program.

No. 37
COUNTRY: SWEDEN
CNS-REF.-ART.: Article 15
PAGE OF REPORT: 117
CHAPTER OF NAT. REPORT: 3.15.5

ARN requires that whenever possible, radiological protection be achieved using plant's systems rather than operational procedures.

Could you please inform about which measures have been taken to achieve this and which modifications have been made to the plant systems.

The statement is in the frame of Requirement 81: "Design for radiation protection" of IAEA SSR 2/1, Rev.1 by which "Provision shall be made for ensuring that doses to operating personnel at the nuclear power plant will be maintained below the dose limits and will be kept as low as reasonably achievable, and that the relevant dose constraints will be taken into consideration."

Despite that this requirement is applicable for the design of new NPPs, one example for the existing NPPs that shows ARN's commitment in this regard, is the requirement for replacement "stellite-6" in the core material to minimize activation of materials, as far as is reasonably practicable (see 3.18.3.2).

No. 38
COUNTRY: SWEDEN
CNS-REF.-ART.: Article 15
PAGE OF REPORT: 121
CHAPTER OF NAT. REPORT: 3.15.6.3

ON-LINE Dosimetry System Management, avoiding unplanned exposures.

Is the system used mostly as a tool in protection against overexposures or it also used in the optimization of radiation protection?

It is both. The system basically works as real time (on-line) dosimetry tool which, as stated, avoids unplanned exposures. However, other solutions (hardware & software) can be included to expand the RP service provided, such as, on-line air contamination measurement and real time IP video streaming (both used as RP optimization tool). At Embalse NPP, we have used the system as a real time dosimetry along with the video streaming, and we are currently in the process of testing the air contamination additions.

No. 39
COUNTRY: SWEDEN
CNS-REF.-ART.: Article 15
PAGE OF REPORT: 121
CHAPTER OF NAT. REPORT: 3.15.6.2

Dose reduction is reported for scheduled outages due to e.g. better control and monitoring of the personnel individual dose and use of tele-dosimetry.

Was there any individual dose planning carried out before the work was performed? Are there requirements for individual dose targets, linked to the practical execution? Was a specific dose rate survey program carried out before the work?

1) The dosimetric planning of the task was carried out taking into account the dose rates in workplace, the time and the amount required, and based on that the collective dose was estimated.
2) There are individual annual ALARA restrictions of 15 mSv for plant personnel and 18 mSv for hired staff.
3) Yes, it was done.

No. 40
COUNTRY: SWEDEN
CNS-REF.-ART.: Article 15
PAGE OF REPORT: 118
CHAPTER OF NAT. REPORT: Table 3.15.10 and Table 3.15.11

The average effective dose at "CNA I" and "CNE" plants is roughly 2 mSv/year; while at "CNA II" its below 1 mSv/year.

What is the reason for the difference between the plants? Please, could you inform about the maximum individual doses at the plants?

The main reason for the difference in the average effective dose between the plants is the operating life time of every plant. As known all Argentinian NPP has a D_2O as coolant and moderator, which in course of time is being more activated. Also with the pass of time are being more contaminated different systems and components in plants. In case of CNA I this time is more than 40 years, in CNE more than 30 years. Of course, despite of the radiological measures, these activation and contamination are reflected in the effective dose of maintenance and operation personal. In case of CNA II the operation time is approximately 5 years and there aren't the same levels of D_2O activation or components contamination as in other plants.

Anyway, there were other tasks done in CNA I and CNE that could contribute to increase the average effective dose. For example, in CNA I, the intervention of the guide tube of the control bar (reactor enclosure), in CNE, the feeders repairmen of channel S07 (calandria enclosure).

The maximum individual doses in 2019 were:

Atucha Site (CNA I + CNA II): 17.673 mSv

CNE: 6.316 mSv

No. 41
COUNTRY: SWEDEN
CNS-REF.-ART.: Article 16
PAGE OF REPORT: 123
CHAPTER OF NAT. REPORT: 3.16.2

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.

Here automatic countermeasures are mentioned. Could you please explain what does the automatic countermeasures mean and consist of?

Automatic countermeasures are protective measures for the population which has to be implemented rapidly in order to reduce the risk of severe effects on the population's health. The established criterion is, except in extreme situations, that automatic countermeasures are applied before the release of radioactive material starts.

No. 42
COUNTRY: SWEDEN
CNS-REF.-ART.: Article 17
PAGE OF REPORT: 139
CHAPTER OF NAT. REPORT: 3.17.2.3.2.1.1: Earthquakes - CNA I

"The possible seismically induced internal flooding was analysed during the systems walk downs" at CNA I.

Can anything be said about the results of these analyses and why they were limited to internal flooding compared to the ones for the CNA II and CNE sites which also included external flooding?

Liquid levels released per site and potential flood levels were calculated. The affected safety components were identified and actions were taken, such as:

- design, construction and installation of non-return mechanisms in different enclosures
- replacement of normal doors with watertight doors
- clapper placement in different enclosures
- placement of qualified seals (flame retardant, fireproof, waterproof and radiation resistant) in certain slab and wall passages
- placement of level sensors
- reconditioning and / or replacement of emergency door (to provide watertight sealing) that connects Level -6.50 m of Auxiliary Building with Manoeuvre Building, to isolate the controlled zone from floods that come from outside.

In addition, an abnormal event instruction is available to diagnose an internal flood event and take measures to mitigate its impact.

CNA I and CNA II are in the same site and external flooding was analysed for the site. (Please see paragraph 3.17.2.3.2.1.2 of the report).

No. 43
COUNTRY: SWEDEN
CNS-REF.-ART.: Article 17
PAGE OF REPORT: 139
CHAPTER OF NAT. REPORT: 3.17.2.3.2.1.1: Earthquakes - CNA II

Some easy fixes have already been solved and others will be implemented in the next scheduled outage in 2017.

Were the planned improvements implemented in 2017 and which were they?

The remaining improvements were made between 2017 and 2018 during outages. Furthermore, some of them were solved during Normal Operation. They were:

JFC44AA201: Solenoid operated valve. Solved during scheduled outage 2017. Enlarge the upper rack strike where a tube passes from the top of the valve.

KAA20AA006: Motor operated valve. Solved during scheduled outage 2018. Thinning of the Wall with which it is in contact.

KAA20AA007: Motor operated valve. Solved during scheduled outage 2018. Turn the valve away from the Wall, to give it a minimum of 1" of separation, as the top can interact with the wall.

BTF/BTD Batteries bank: Solved during Normal Operation in 2016. Mobile cranes were accommodated to avoid a possible fall in the batteries racks in case of earthquakes.

No. 44
COUNTRY: SWEDEN
CNS-REF.-ART.: Article 17
PAGE OF REPORT: 139
CHAPTER OF NAT. REPORT: 3.17.2.3.2.1.1: Earthquakes - CNA II

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.

Can anything be said about the actions needed?

Liquid levels released per site and potential flood levels were calculated. The affected safety components were identified and actions were taken, such as:

- design, construction and installation of non-return mechanisms in different enclosures
- replacement of normal doors with watertight doors
- clapper placement in different enclosures
- placement of qualified seals (flame retardant, fireproof, waterproof and radiation resistant) in certain slab and wall passages
- placement of level sensors
- reconditioning and / or replacement of emergency door (to provide watertight sealing) that connects Level -6.50 m of Auxiliary Building with Manoeuvre Building, to isolate the controlled zone from floods that come from outside.

An abnormal event instruction is available to diagnose an internal flood event and take measures to mitigate its impact.

In addition, there are actions in the Operations Manual that address external flooding, regardless of the event that originates it. (Instruction T17 for CNAUI, 3.3.3 for CNAUII).

No. 45
COUNTRY: SWEDEN
CNS-REF.-ART.: Article 17
PAGE OF REPORT: 139
CHAPTER OF NAT. REPORT: 3.17.2.3.2.1.1: Earthquakes - CNA II

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.

Can any examples of improvements be given?

Examples of improvements:

JFC44AA201: Solenoid operated valve. Solved during scheduled outage 2017. Enlargement of the upper rack strike where a tube passes from the top of the valve.

KAA20AA006: Motor operated valve. Solved during scheduled outage 2018. Thinning of the Wall with which it is in contact.

KAA20AA007: Motor operated valve. Solved during scheduled outage 2018. Turn the valve away from the Wall, to give it a minimum of 1” of separation, as the top can interact with the wall.

BTF/BTD Batteries bank: Solved during Normal Operation in 2016. Mobile cranes were accommodated to avoid a possible fall in the batteries racks in case of earthquakes.

No. 46
COUNTRY: SWEDEN
CNS-REF.-ART.: Article 17
PAGE OF REPORT: 144
CHAPTER OF NAT. REPORT: 3.17.2.3.2.2.1.

A 24-inch rupture disc (RD) assembly shall be installed on the top of the existing calandria vault inspection port […]. It is foreseen to be implemented by the end of 2017.

Has the rupture disc been installed by now or has this been postponed for any reason?

In order to preserve the structural integrity of the Calandria vault, an additional rupture disk (RD) was installed during the CNE Life Extension. This new disk was added to the existing ones that belong to the Extreme Shielding Cooling System.

With this design modification, the pressure relief capacity of the Calandria vault is ensured during a severe accident. The existing disks are 6-inch and the new is 24-inch.

No. 47
COUNTRY: SWEDEN
CNS-REF.-ART.: Article 17
PAGE OF REPORT: 137-139
CHAPTER OF NAT. REPORT: 3.17.2.3.2.1.1

For CNA I, CNA II and CNE it is pointed out that seismically induced flooding has been analysed. Have analyses on seismically induced fire been performed as well?

For CNA UI the revision 1 of the Internal Fire Analysis (Phase 2 PSA Level 1) was developed and submitted to the regulatory body by incorporating the Second Heat Sink, the Emergency Power Supply System and the fourth UK pump, and based on PSA Revision 5.

CNA UII is developing the Internal Fire PSA, which will be finished by midst 2021.

For CNE, analysis of seismically induced fire was performed as part of the PSA based Seismic Margin Assessment. The Embalse SSEL was reviewed to identify the sources of seismically induced fire. Fire scenarios were developed for each source by identifying the sources’ location, paths of propagation, targets and damaged components. The Embalse seismically induced fire sources were identified to be only NSQ fuel tanks. In general, the seismic fire interaction sources may lead to leak of oil due to rupture of the tanks as a result of the earthquake event. In all cases, an electric spark was assumed to be the ignition source.

The result of the assessment was that all seismic induced fire scenarios could be screened out and no further assessment was required in the level 1 and 2 PSA based SMA.

No. 48
COUNTRY: SWEDEN
CNS-REF.-ART.: Article 18
PAGE OF REPORT: 150-155
CHAPTER OF NAT. REPORT: 3.18.3.1/ 3.18.3.3

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 (p. 150). 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 (p.155).

Are the requirements related to man-machine interface in Regulatory Standard AR 3.4.1 different depending on when a reactor was designed? How does CNA II comply with the Regulatory Standard AR 3.4.1?

Although the standard AR 3.4.1 does not take into account for the establishment of different requirements the moment at which the reactor was designed, it is a fact that Argentina regulations are frequently updated (or supplemented) in order to maintain the requirements according to the state of art and aligned with international standards. This may result in the consideration of additional regulatory requirements that are required for the design of new reactors and in some cases may generate specific requirements that imply design modifications for the existing nuclear reactors.

The AR 3.4.1 standard basically establishes the following requirements related to the man-machine interface:

- The reactor protection system must initiate its actions automatically.
- Operator action is not necessary for a required period of time after activation.
- The operator can initiate protection actions but cannot activate the operation of the reactor protection system.
- The operator must have the indication of the status of all protection actions.

In CNA II, these criteria are fulfilled and all protection actions performed by the reactor protection system are carried out automatically by monitoring the safety variables representative of the different design basis accidents and without the operator intervention. Similarly, the concept of priority control is used to identify that reactor protection commands have priority over operational commands (both from operating systems and manuals performed by the operator) and thus, avoiding the operator interruption of protective actions. Finally, the status of all protection actions can be displayed on the reactor protection system panel located in the main control room and in the emergency control room.

No. 49
COUNTRY: SWEDEN
CNS-REF.-ART.: Article 18
PAGE OF REPORT: 150
CHAPTER OF NAT. REPORT: 3.18.3.1.1

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.**

In light of VDNS (principle 2), should this be understood as no decision has yet been made whether these changes will be implemented or not?

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 were analysed and extra efforts had to be made to adapt codes to the Atucha reactors. 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). In the past years, preliminary results were obtained with RELAP5 / SCDAP. These calculations were followed by more complex analysis with ANSYS / CFD code, performed for CNA II NPP. The results of these analyses showed that the countermeasure is not successful in a scenario of LOCA in the moderator circuit with failure of safety injection system or in a SBO scenario. Based on these results, it was decided to rule out this countermeasure for Unit II and Unit I. It should be noted that the results for Unit II are extrapolable to Unit I in this case. As it was mentioned a parallel course of analysis has been started, to assess possible stabilization of molten material inside sump, to avoid an early containment breach, so as to decrease consequences in public as far as reasonable achievable. This task is being performed as part of phase B of CNA I LTO. This project is part of the Conceptual Improvement Plan that will be presented for consideration to ARN in March 2020 (see section 3.6.5.2).

Regarding the venting filtered containment system it is also planned to be implemented as part of phase B of CNA I LTO.

No. 50
COUNTRY: SWEDEN
CNS-REF.-ART.: Article 18
PAGE OF REPORT: 154
CHAPTER OF NAT. REPORT: 3.18.3.2.2

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 […].

What is the status of implementing the alternative power sources and the venting filtered containment system?

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 were analysed and 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). 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 showed that the countermeasure is not successful in a scenario of LOCA in the moderator circuit with failure of safety injection system or in a SBO scenario. Based on these results, it was decided to rule out this countermeasure for Unit II and Unit I. As it was mentioned a parallel course of analysis has been started, to assess possible stabilization of molten material inside sump, to avoid an early containment breach, so as to decrease consequences in public as far as reasonable achievable.

The filtered venting containment system for CNA II is planned to be implemented at the same time with CNA I in order to take the advantages of carrying out similar projects together. It implementation is foreseen for phase B of CNA I LTO.

Regarding alternative power sources, an electrical interconnection between normal busbars of Unit I and Unit II is available. It should be noted that Unit 1 is connected to 220 kV line and Unit 2 is connected to 500 kV line and both units have 132 kV line as a back up line. A new mobile diesel generator will be available on site in May 2020.

No. 51
COUNTRY: SWEDEN
CNS-REF.-ART.: Article 11
PAGE OF REPORT: 71
CHAPTER OF NAT. REPORT: 3.11.2.3

The total personnel has decreased from 3435 to 2996 from 2017-2018.
Will this continue? If so what kind of proactive measures are taken?

Currently there are 200 people at NA-SA who will be retired in the next two years. The company is working proactively to have different scenarios for a medium-term staffing plan, which is related to the annual budget according to the current budget of the company (with a 3-year horizon).

No. 52
COUNTRY: SWEDEN
CNS-REF.-ART.: General
PAGE OF REPORT: 4
CHAPTER OF NAT. REPORT: 1.4.1

On page 4 it is written: "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 tests to the Licensees.

Since FORO consists of 10 countries but only four countries have nuclear reactors, on what basis was this consensus reached? Were the technical issues of the stress tests agreement different from the European (of which Spain also took part)?

As it is stated in section 1.4.1, member states having NPPs of the FORO decided to conduct a stress test in each one of their NPPs. The consensus refers to the agreement reached by these member states in relation of performing a stress test with similar scope and content to the one implemented by the Western European Nuclear Regulators Association (WENRA). The technical issues of the stress test agreement didn't have any difference with respect to the European one.

No. 53
COUNTRY: SWEDEN
CNS-REF.-ART.: General
PAGE OF REPORT: 6
CHAPTER OF NAT. REPORT: 1.4.1.1

Bullet 9 (on page 6) reads; Installation of an additional (fourth) pump to the river Water Cooling Ensured System (UK) (CNA I). What does UK refer to in this context?

In CNA I and based on German rules, it was used an identification system for the plant systems-structures-components called KKS. In the frame of this KKS, the letter U refers to mechanical group of conventional secondary installations and UK refers to secondary secured cooling system.

No. 54
COUNTRY: SWEDEN
CNS-REF.-ART.: GENERAL
PAGE OF REPORT: 9
CHAPTER OF NAT. REPORT: 1.4.2.1

On page 9 it is written; "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 by to ARN".

What is the legal status of such an MOU in Argentina? What would the implications be if ARN or NA-SA would not honour the MOU? In what way can the safety level be "fixed" or "outlined" at this early stage of the project?

The current licensing process of NPPs in Argentina starts with the construction stage and there is no formal arrangement in the regulation for any earlier regulatory involvement.

Currently, ARN is updating their standards including the one corresponding to licensing process for nuclear installations. As part of this activity, it is foreseen the adoption of “pre-licensing” process by which the design of the proposed NPP is reviewed against the AR’s standards as well as IAEA SSR 2/1 Rev. 1 and IAEA SSG-30.

In the meantime, ARN developed the MOU as a formal arrangement to frame the “pre-licensing” activities and the regulatory expectations. Because MOU was signed by the Board of Director of ARN, it became a formal regulatory requirement of mandatory fulfilment by the applicant.

In the case that NA-SA (the applicant) doesn´t honour the MOU, ARN has the right to decide different actions ranking from the issuance of additional regulatory requirements up to decline the licensibility of such NPP, depending of the safety implication of the deviation from the MOU.

During this early stage of the project, a high level approach is used for reviewing the safety level of the proposed plant through the fulfilment of requirements as stated in IAEA SSR 2/1 Rev. 1 and the so called “integrality concept”. By this concept, 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.

No. 55
COUNTRY: SWEDEN
CNS-REF.-ART.: General
PAGE OF REPORT: 9
CHAPTER OF NAT. REPORT: 1.4.2.2.1

It is stated on page 9 that; "PSR is used in Argentina for justification and development of the analysis of the minimum modifications to be done for safe continued operation". How does ARN enhance safety and optimise radiation protection if only minimum modifications are required? How does this objective compare with the VDNS, objective 2, of implementation of reasonably practicable safety improvements are to be implemented in a timely manner?

Please, understand the wording “minimum” in this context as following: PSR is not the only source driving the identification of improvement/enhancement activities for safe continued operation.

See section 3.6.3, where it is stated that safety improvements are driven by different approaches.

No. 56
COUNTRY: SWEDEN
CNS-REF.-ART.: General
PAGE OF REPORT: 12
CHAPTER OF NAT. REPORT: 2.2

On page 12 it is stated: "Regarding the recommendations, the most relevant are related to the methodology for scope setting for assessments of SSCs for LTO and implementation of a comprehensive equipment qualification programme.” Thank you for sharing the information about recommendations and suggestions. Should "most relevant" not read "most important" in this section? Were not all the recommendations relevant?

The recommendations were all relevant and both parties: NA-SA and ARN, appreciate them.

Correct, “most relevant” not read “most important”.

No. 57
COUNTRY: SWEDEN
CNS-REF.-ART.: General
PAGE OF REPORT: 15
CHAPTER OF NAT. REPORT: 2.9

It is stated on page 15 that: "The goal of the PRACS is to create a bridge between the concepts of Nuclear Safety Culture and actual performance in the installation". Due to the very definition of nuclear safety culture, the actual performance in the installation is influenced by the prevailing safety culture. Please explain what is meant by performing a bridge in this context?

Safety culture in the first instance may seem in its definition as an abstract term. What is intended with the PRACS is to provide a practical implementation framework to reinforce the safety culture, by strengthening specific issues (Fire Protection, Emergency Preparedness, Equipment Reliability, Human Performance, Operational Decision Making, Indicator Management, among others), through the definition of persons responsible (coordinators), in the different NA-SA sites, who lead these issues, which must implement a specific methodology to carry out transversal objectives to the organization, through the fulfilment of specific actions. The term “bridge” refers to how a definition of safety culture that speaks of "values" makes a concrete transduction to specific actions.

No. 58
COUNTRY: SWEDEN
CNS-REF.-ART.: Article 6
PAGE OF REPORT: 19
CHAPTER OF NAT. REPORT: 3.6.3.1.2

It is reported (page 19); "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…the results of these 47 reports were 374 recommendations categorized as non-critical for safe long term operation". In the next section it is written: "In addition, CNA I, CNA II and CNE are improving their current Ageing Management Program using the latest table of International Ageing Lessons Learned (IGALL - AMPs)...it can be pointed out the need of 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".

Was the original LTO review performed against "obsolete standards" and if so, does this explain why no critical recommendations for safe long term operation of the reactor constructed between 1968 and 1974 was found?

LTO in Argentina is justified using the Periodic Safety Review. Thus, review of the safety factors is performed against modern standards, including the particular ones for ageing.

Recommendations were categorized as non- critical for safe LTO as all of them are feasible to be managed assuring a continued safe operation.

No. 59
COUNTRY: SWEDEN
CNS-REF.-ART.: Article 6
PAGE OF REPORT: 20
CHAPTER OF NAT. REPORT: 3.6.4.1.1

It is mentioned (page 20) that; "During the interim storage the spent fuel must maintain the same structure and integrity as those one that have never been deposited dry". Is not the general view that dry storage of spent fuel (in an inert environment in order to counteract corrosion) is less problematic than wet storage? What is your view on this issue?

Yes, we agree that dry storage of spent fuel is less problematic than wet storage.

But, in order to understand the situation and context in CNA I (Nuclear Power Plant Atucha I) in relation with the storage of spent fuel, we will describe some design details of the Plant:

a) There are originally 2 (two) spent fuel bay buildings in CNA I (CP1 y CP2).
b) Their capacities are: CP1 – 3,240 positions for spent fuels; CP2 – 8,304 positions for spent fuels. At 31/12/2019, there were occupied 10,701 of these positions (approx. 93%).
c) The normal operation in CNA I is supported by a regularly refill of fuel. The average is 0.7 spent fuel/Day.

In order to support the normal operation of CNA I in the future, it was decided to construct a dry storage facility of Spent Fuel Elements with a 2,754 spent fuel capacity. By design, this facility will be integrated to the existing building of the spent fuel bay building with a possible reversible moving process of fuel elements. It will permit the spent fuel elements store dry, but in case of necessity, they can go again through the wet storage, for example in case they will go to the final disposal.

No. 60
COUNTRY: SWEDEN
CNS-REF.-ART.: Article 7
PAGE OF REPORT: 40
CHAPTER OF NAT. REPORT: 3.7.3.3

It is stated (page 40) that; "Act No. 24,804, entitles the ARN to carry out with such inspections and regulatory review and assessments, performed by its personnel, such as:..." and then it is listed "routine planned inspections", "special inspections, including reactive inspections", "safety assessments" and "regulatory audits". Does the Act detail which type of inspections and assessment the ARN can perform (e.g. unannounced inspections?) or is this decided by the authority within a more general mandate? What is a meant by "special circumstance" besides the abnormal events?

The Act states that ARN has to carry out inspections and the frequency of them has to be determined by ARN. It is also said in the Act that ARN has the power to define the access regime for their inspectors.

There is no additional detail in the Act regarding the inspections.

Special circumstances is referring to, besides the abnormal events, those situations arisen from the non-compliance of the terms and conditions of the corresponding Licence. It is a broader approach compared with the only consideration of abnormal events.

No. 61
COUNTRY: SWEDEN
CNS-REF.-ART.: Article 7
PAGE OF REPORT: 40
CHAPTER OF NAT. REPORT: 3.7.3.4

The text on page 40 describes that "ARN, in certain cases, can issue a recommendation to the licensee. A recommendation is a demand that differs from a requirement in that the licensee has certain flexibility to accomplish it by means of alternative ways which ensure at least the same result required by the recommendation". Since on page 33-34 it is argued that the regulatory standards are non-prescriptive, the licensee's responsibility goes beyond the mere compliance with requirements, and that ARN mostly work with a performance based approach, what would such a recommendation consist of? Does it recommend a certain outcome or means to achieve this outcome? Could you please give examples of such recommendations?

According to ARN's management system the definition of recommendation is the following:

"Recommendation: It is a regulatory requirement for whose compliance the Responsible Entity has some flexibility, being able to adopt alternative solutions (for example different engineering solutions) to ensure, at a minimum, the same result required by the recommendation. These alternative solutions must be proposed to ARN for evaluation".

No. 62
COUNTRY: SWEDEN
CNS-REF.-ART.: Article 8
PAGE OF REPORT: 43
CHAPTER OF NAT. REPORT: 3.8.1

On page 43 it is written that the “Law No. 25,018/98 sets provisions that involve the ARN in the management of Radioactive Wastes and that it states that ARN must approve the acceptance criteria, the transfer conditions and the radioactive waste (irradiated fuel) transference procedures.” What does it mean that ARN approves these criteria, conditions and procedures (CCPs)? Is any safety responsibility taken over by the authority? May other CCPs be used if they fulfill the requirements of the laws and standards?

ARN, as Regulatory Body, has the competence to authorize, among other things, the transfer and storage of radioactive waste and irradiated fuel, for which it evaluates the specific conditions (criteria and procedures) if any, proposed by CNEA for such purpose, and approves if applicable. The responsibility of the tasks belongs to the operator.

No. 63
COUNTRY: SWEDEN
CNS-REF.-ART.: Article 8
PAGE OF REPORT: 44
CHAPTER OF NAT. REPORT: 3.8.2

UCE responsibility listed at page 44: Institutional knowledge management. Could you please tell us more about how ARN works with institutional management knowledge?

The Education and Training Unit, in partnership with some relevant sectors, is developing a general training plan. Knowledge management activities are included in this general strategy. Regarding the institutional knowledge management, among other activities, it is expected to build an active database (particularly, it will contain the data linked to the E&T activities carried out by the staff), a systematic appraisal the material produced by the institution and implement the application of SARCON tool provided by the IAEA.

No. 64
COUNTRY: SWEDEN
CNS-REF.-ART.: Article 8
PAGE OF REPORT: 46
CHAPTER OF NAT. REPORT: 3.8.3

It is reported on page 46 that activities are on-going in order to define ARN job profiles. Furthermore, the HRD division will develop a training plan for long and medium term. Could you please share your experiences in this important task? Have you also mapped the available staff competences?

Now there is a document that describes the basic profiles and training requirements for staff. This document lays the basis for developing the training plan but, in order to develop this planning, it is necessary to know the real skills of the staff and the existing gaps. Therefore, the ongoing activities are the analysis of the skills gaps defined to map them and develop the training plan.

No. 65
COUNTRY: SWEDEN
CNS-REF.-ART.: Article 8
PAGE OF REPORT: 48
CHAPTER OF NAT. REPORT: 3.8.3.1.2

It is reported on page 48 that one of the main changes affecting NPPs control was in 2015 when inspections and evaluations related to radiological safety were transferred to the Radiological Protection in Facilities and Practices Division, in coordination but independent from LCRND. Could you please tell us more about the reasons behind this transfer?

Historically, the regulatory activities in radiation safety field were performed by professional belonging to the former division of "Radiological Protection in Facilities and Practices" in response to a request of activity identified by "Licensing and Control of Nuclear Reactors". In 2015, instead of working on a request basis, the Board of Directors decided to allocate the responsibility for radiation safety and radiation protection to the "Radiological Protection in Facilities and Practices". So, the main reason is the direct allocation of responsibility to the division where the professionals belong to.

No. 66
COUNTRY: AUSTRIA
CNS-REF.-ART.: Article 7
PAGE OF REPORT: 31
CHAPTER OF NAT. REPORT: 3.7

Please specify were the requirements for LTO are laid down and elaborate them.

There is no indication were the legal requirements for LTO are elaborated. It is unclear if they are part of a law, decree, or regulatory guides.

Up to date there is no AR standard defining the regulatory requirements and expectations for LTO. However, by regulatory practice all the requirements for a safe LTO are developed in formal letters which became as mandatory fulfilment. See for more detail the section 3.6.5.2 of the National Report.

In the case of Atucha I these requirements 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.

No. 67
COUNTRY: AUSTRIA
CNS-REF.-ART.: Article 7
PAGE OF REPORT: 35
CHAPTER OF NAT. REPORT: 3.7.2.3.1

On page 35 it is stated that the "ARN adopted, more than three decades ago, a probabilistic criterion for defining reference levels of acceptable risk". Do you plan to include the VDNS Principle 1 in your NORMA AR 3.1.3 "Criterios radiológicos relativos a accidentes en reactores nucleares de potencia (Rev. 2)" document? Please define the "Grupo Crítico" in NORMA AR 3.1.3.

The Argentinian legislation is based on probabilistic criteria for defining reference levels of acceptable risk. Those reference levels are elaborated in AR 3.1.3. "Criterios radiológicos relativos a accidentes en reactores nucleares de potencia (Rev. 2)". Within this document the levels of acceptability and non-acceptability based on the probability and the dose for the public are elaborated. The curve does not reflect the VDNS Principle 1 regarding L&ERFs to be practically eliminated.

Currently there is a project for the review of all the Argentine standards, and the Vienna Declaration on Nuclear Safety (VDNS) will be included. The critical group stated in the Standard AR 3.1.3 is nowadays interpreted as a hypothetical person who lives in the surroundings of the Nuclear Power Plant and that has the highest risk (mainly due to his location taking into account the dominant wind direction and the distance to the NPP). The risk is measured as the probability of a fatality (or having a severe consequence) due to an irradiation following a nuclear accident.

No. 68
COUNTRY: AUSTRIA
CNS-REF.-ART.: Article 7
PAGE OF REPORT: 35
CHAPTER OF NAT. REPORT: 3.7.2.3.1

According the information provided on page 35 "Regulatory Standards are not prescriptive but of compliance with safety objectives (performance)". From the safety objectives the acceptance criteria are derived for each NPP. Please provide a comparison of the acceptance criteria for your three NPPs.

It is not easy to provide a comparison of the acceptance criteria for the different NPPs. For example, some deterministic acceptance criteria are identified from functional capacity, reliability and robustness derived from the safety classification of systems, structures and components, which in turn is based on the Safety Analysis demonstrating the functional safety of a design.

No. 69
COUNTRY: AUSTRIA
CNS-REF.-ART.: Article 15
PAGE OF REPORT: 117
CHAPTER OF NAT. REPORT: 3.15.5.1

Chapter 15.5.1 provides an overview on manSv and average dose at the Argentinian NPPs. Can you provide information on the maximum dose a person received in the respective years?

An average dose of >2mSv is rather high in comparison for pressurized water reactors and with regard to other countries.

Maximum individual annual dose for CNAUI

	NA-SA	Contractor
2016	14.89 mSv	17.37 mSv
2017	14.85 mSv	18.90 mSv
2018	15.28 mSv	16.47 mSv

Maximum individual annual dose for CNAUII

	NA-SA	Contractor
2016	2.55 mSv	1.52 mSv
2017	9.66 mSv	9.94 mSv
2018	7.13 mSv	5.85 mSv

Maximum individual annual dose for CNE

2016	20.44 mSv
2017	19.68 mSv
2018	19.33 mSv

No. 70
COUNTRY: AUSTRIA
CNS-REF.-ART.: Article 17
PAGE OF REPORT: 133
CHAPTER OF NAT. REPORT: 3.17.2.1

“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"

What are the standards included in AR 10.10.1? Is IAEA SSR-1 Site Evaluation for Nuclear Installations considered? To what extent are possible effects of climate change considered?

The standard on which the AR 10.10.1 was based is the IAEA NS-R-3, Rev.1 - Site Evaluation for Nuclear Installations.

AR 10.10.1, as well as the revised NS-R-3, contains requirements related to:

- The potential occurrence of events in combination;
- Establishing levels of hazard for the design basis for the installation and their associated uncertainties;
- Multiple facilities at a single site;
- Monitoring of hazards and periodic review of site specific hazards.

There are no specific requirements related to the consideration of climate change.

No. 71
COUNTRY: AUSTRIA
CNS-REF.-ART.: Article 17
PAGE OF REPORT: 137
CHAPTER OF NAT. REPORT: 3.17.2.3.2.1.1

The report states that “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".

What was the initial Design Basis Earthquake for CNA I? Is the seismic safety assessment completed for CNA I? If so, what are the final findings and resulting improvements? What were the results regarding possible seismically induced internal flooding?

The design of CNA I, consistent with the criteria and requirements established in the 1960s for nuclear power plants located at sites of low seismicity, did not contemplate seismic loads.

In the context of Fukushima, following WANO recommendations and ARN requirements the evaluation of the safety against earthquakes and associated induced hazards, for CNA I was formulated by NA-SA.

The seismic evaluation program consisted of the following phases:

- Phase 1: Scoping Study and Preliminary Plant Walkdown
- Phase 2: Development of the Safe Shutdown Equipment List (SSEL) and System Walkdown
- Phase 3: Structures Seismic Dynamic Response and In-structure Response Spectra Calculations
- Phase 4: Seismic Capability Walkdown – Screening Process
- Phase 5: Detailed Analysis and Evaluation for Seismic Qualification.

Those equipment and / or structures whose seismic capacity was judged below acceptable were classified as outliers, and the solutions were categorized as follows:

- items for which easy fixes were implemented
- items that depended on chatting of relays or contactors, for which studies and tests were performed, in order to qualify them
- items for which HCLPF calculations were performed, and, if necessary, plant modifications were implemented

Regarding internal flooding:

Liquid levels released per site and potential flood levels were calculated. The affected safety components were identified and actions were taken, such as:

- design, construction and installation of non-return mechanisms in different enclosures
- replacement of normal doors with watertight doors
- clapper placement in different enclosures
- placement of qualified seals (flame retardant, fireproof, waterproof and radiation resistant) in certain slab and wall passages
- placement of level sensors
- reconditioning and / or replacement of emergency door (to provide watertight sealing) that connects Level -6.50 m of Auxiliary Building with Manoeuvre Building, to isolate the controlled zone from floods that come from outside.

In addition, an abnormal event instruction is available to diagnose an internal flood event and take measures to mitigate its impact.

No. 72
COUNTRY: AUSTRIA
CNS-REF.-ART.: Article 17
PAGE OF REPORT: 141
CHAPTER OF NAT. REPORT: 3.17.2.3.2.1.3

The report states that some conceivable weaknesses were identified concerning other external hazards and the "licensee decided to implement additional studies to confirm them". Can you elaborate on the weaknesses? Have the studies been completed? Have any hard ware measures been implemented?

Aircraft traffic: According to IAEA Standard NS-G-3.1 ""External Human Induced Events in Site Evaluation for Nuclear Power Plants"" (2002), and the information provided by the Argentine Air Force, both airways located over CNA as the proximity of airports can be ruled out as sources of risk since they are beyond the Screening Distance Value (SDV) recommended for airways and proximity to airports (4 km). In addition, the CNA zone is a prohibited flight zone because the restriction is 3000 feet (914.4 m) and the airways are above that height. The annual probabilities of a plane and helicopter accident in the CNA linear airway are less than 10^{-7}/ year, so according to the Argentine Regulatory Guide ARN 3.1.3 these scenarios are discarded.

Potentially hazardous industrial plants: According to the information provided by the Municipality of Zárate, within this area there are chemical factories, and few of them are dangerous for human health and with the possibility of forming a toxic cloud. All of them are beyond the SDV suggested by the IAEA for explosions and toxic clouds.

No. 73
COUNTRY: AUSTRIA
CNS-REF.-ART.: Article 17
PAGE OF REPORT: 145
CHAPTER OF NAT. REPORT: 3.17.2.3.2.2.3

In chapter 3.17.2.3.2.2.3 a re-evaluation of the risk of tornadoes for CNE is mentioned, but no results / findings are presented. Are there any actions resulting from the evaluation?

As the result of the assessment of risk of tornado for the CNE site, the actions proposed are:

1. Cleanup from outdoor areas, all loose/stored components that could lead to be potential missiles
2. In order to guarantee the 5 key safety functions under the occurrence of a tornado, an assessment was performed and the conclusion was to take reinforcement measures in civil structures in EPS and EWS buildings and the housing where some isolation valves are located.

No. 74
COUNTRY: AUSTRIA
CNS-REF.-ART.: Article 18
PAGE OF REPORT: 148
CHAPTER OF NAT. REPORT: 3.18.2

The report states that for 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". What are the related measures to achieve a robust DiD concept? Which are the main components addressed by these measures? To what extent are DECs considered? What are the related expected probabilities of occurrence and acceptance criteria?

The methodology for assessing the Defence in Depth (DinD) concept in Atucha I was developed based on the IAEA SRS-46, "Assessment of Defence in Depth for Nuclear Power Plants".

According to this document, there a series of safety principles (affecting different levels of DinD) that has to be assessed in order to identify the improvement/corrective measures for achievement a robust DinD concept.

Examples of these safety principles are SP 182: Equipment Qualification, SP 177: dependent failures, SP 221: Containment structure protection, etc.

As per this methodology all systems, structures and components important to safety need to be assessed as a first step.

In the case of Atucha I after doing this, corrective measures like replacement of non-qualified for qualified equipment, protective measures as installation of additional piping supports or re-routing/re-location of some systems, among others activities were and are being performed.

Design extension conditions without significant fuel degradation as well as design extension conditions with core melt, are considered in the above mentioned methodology.

No. 75
COUNTRY: AUSTRIA
CNS-REF.-ART.: Article 18
PAGE OF REPORT: 148
CHAPTER OF NAT. REPORT: 3.18.2

It is stated that 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". A greatly simplified design might contradict an enhanced Defence in Depth concept. This should be elaborated.

Can you elaborate on the enhanced Defence in Depth concept of the CAREM? How is the VDNS Principle 1 reflected in the license requirements?

According to the international trend and the recommendations of the IAEA, the design of CAREM-25 incorporates innovative concepts. Special emphasis has been placed on the adequate internalization of the Defense in Depth principle with the objective of fulfilling the Fundamental Safety Functions.

As mentioned in section 2.4, 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 sublevel 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.

Regarding to the currently construction license, the requirements derived from the regulatory standards are maintained, adding the requirements established in the AUSC (section 3.18.3.4.3.) related to some systems (of regulatory interest) that operate at level 3B of DiD. Those requirements applies to design, construction, assembly and testing stages of 2nd Protection System, 2nd (Diverse) Shutdown System, depressurization valves of RPV and Containment.

No. 76
COUNTRY: AUSTRIA
CNS-REF.-ART.: Article 18
PAGE OF REPORT: 149
CHAPTER OF NAT. REPORT: 3.18.3

On page 149 it is stated that some regulatory standards were not immediately applied at CNE and CNA I. When and to what extent where the ARs applied to the older NPPSs?

Since 2003, in Argentina, the Operating Licence is granted for a time period of ten calendar years. The regulatory requisite in order to renew the licence, is the submission and approval of a Periodic Safety Review (PSR). As part of this PSR, the safety factor 1 - Design, includes the comparison of the plant against modern standards from the country, as well as from the technology's country.

So, the AR are not exactly "applied" to older plants but the plant is compared against the ARs and if there are some gaps, corrective measures have to be identified and implemented, as far as practicable and suitable.

No. 77
COUNTRY: FRANCE
CNS-REF.-ART.: Article 8
PAGE OF REPORT: 44
CHAPTER OF NAT. REPORT: 3.8.2

Could Argentina give information about the Nuclear Regulatory Authority ARN's resources to inform the public?

ARN has a Division of Communication responsible for promoting the ARN's institutional image among the stakeholders through strengthening internal and external communications. The Division reports to Board of Directors (see Figure 3.8.1 – ARN Organization Chart), with a team of 10 mixed-skill professionals. The functions of media relations and monitoring, online communications, including social media, and visual design are performed by the ARN's Communication staff.

No. 78
COUNTRY: FRANCE
CNS-REF.-ART.: Article 8
PAGE OF REPORT: 43
CHAPTER OF NAT. REPORT: 3.8.2

Could Argentina precise if the three members of the Nuclear Regulatory Authority ARN's Board of Directors are nominated by the President of the Nation or if their nominations are approved through public hearings?

Article 18 of the "National Law of Nuclear Activity", Law Nº 24,804, establishes that the Board of Directors of ARN is nominated by the President of the Nation.

No. 79
COUNTRY: FRANCE
CNS-REF.-ART.: Article 10
PAGE OF REPORT: 15
CHAPTER OF NAT. REPORT: 2.9

Could Argentina describe activities and practices (related to organization, self-assessment, training, etc.) implemented by the nuclear regulatory authority ARN for developing and maintaining the safety culture within the regulatory body and ensuring a "common understanding of the safety culture" between the regulator and the licensee? In addition, does the regulator evaluate their effect on oversight of the licensee's safety culture?

The creation of CNEA in 1950s with the aim to coordinate and promote research in nuclear activities and to control them led to an early awareness of radiological safety issues related to such activities. There was a relevant participation of argentine experts in the development of the concept of "Safety Culture", since the first INSAG meetings working in the analysis of the Chernobyl accident. Since then, the Safety Culture concept was consolidated in the area of nuclear safety, and in the oversight of reactors.

An assessment by the Senior Management on ARN's Safety Culture has found that the following are elements providing strength:

- ARN personnel features a sound radiological safety-awareness based on a well settled common understanding of the fundamentals of radiological protection. Regarding Nuclear Safety, the personnel related to the regulation of Nuclear Facilities has developed an effective competence, based on the theoretical understanding of the conceptual structure and on the practice of relevant regulatory processes successfully performed.
- The concepts of safety culture are generally known throughout ARN organizational structure, and many of its elements are consolidated.
- Within each management department there is an organizational working culture that supports and encourages trust, collaboration and communication.
- Every agent is ensured to be entitled to report of problems relating to technical, human and organizational factors. There is a practice on the acknowledgement of problems, and on the reporting of the actions taken.
- ARN working culture is strong on a questioning and learning attitudes at all levels.
- ARN regulatory control has historically included a systemic approach to enhancing safety in all the facilities and practices under regulation. The trend to the explanation of requisites and teaching attitude on radiological protection towards the responsible of regulated facilities has been a feature of ARN for decades.

Nowadays the concept of Safety Culture is fostered in ARN through several means:

- The Induction Course implemented for all ARN personnel included modules on Radiological Safety (concepts on radiation, biological effects and radioprotection) and Nuclear Safety (concepts of safety by design, demonstration of safety and licensing basis). This allowed to overcome the lack of a systematic induction, and to extend the safety awareness providing a basis to develop Safety Culture as an organizational working culture.
- As part of the documentation relevant for safety and the re-training of personnel at nuclear reactors, Nuclear Fuel Cycle Facilities, and radiative applications.
- Departmental managers are proactive in this regard and foster meetings with inspectors and technical personnel to enhance their safety culture.
- Workshops with users of radiation sources, to enhance their knowledge on safety and their Safety Culture.

No. 80
COUNTRY: FRANCE
CNS-REF.-ART.: Article 11
PAGE OF REPORT: 71
CHAPTER OF NAT. REPORT: 3.11.2.3

Argentina has identified the risks of knowledge and competence loss related to the departures of staff for the licensee NA-SA. Could Argentina indicate if it has taken provisions to maintain the level of knowledge through hiring and knowledge transfer provisions or methods for identifying critical knowledge gaps?

Yes, the licensee continues to work hard to reduce the gaps in critical knowledge. NA-SA has been working on a position-by-position survey to determine the status of these positions during 2018. The study allowed making an effective projection of the potential retirements of personnel that the organization would have in the next 3 years. For each position, an evaluation was made of the impact of the loss of knowledge and experience that comes with potential resignations and potential successors were identified that would replace the vacancies originated by the retirements.

With respect to actions to retain critical knowledge in the organisation, the Training Departments began working on the development of training plans based on the SAT methodology to establish guidelines for the training process of each of the critical positions in the organisation. Together with the fieldwork and the preparation of plans, the training and knowledge module was developed in the Peoplenet management system to load the training plans and closely monitor the training process of the people who hold these positions. It should be noted that the degree of development of the training plans is not the same at the sites. It is expected that by the end of 2020 all sites will have developed plans and uploaded them into the system for subsequent monitoring.

The above mentioned action aims at ordering, retaining and enabling the transfer of knowledge in those critical positions of the organisation. Further more the succession plan is continuously updated. This plan includes the filling of critical positions, the identification of potential successors, planning for potential retirements, in addition to the systematic review of training plans.

No. 81
COUNTRY: FRANCE
CNS-REF.-ART.: Article 13
PAGE OF REPORT: 82-87
CHAPTER OF NAT. REPORT: 3.13

Could Argentina precise procedures and guidance to manage detection of non-conforming, counterfeit, suspect or fraudulent items received from suppliers before they are installed in the plant? Could Argentina precise the inspection program focusing on preventing and detecting the incorporation of non-conforming, counterfeit, suspicious and fraudulent items?

On the basis of a graded approach to safety, a reception committee in each plant is responsible for verifying that the technical specifications of the received items comply with those stated in the purchase order. Also, the items are stocked in controlled conditions according to the specifications. Before installation in the plant, items which do not comply with defined standards are clearly identified and segregated to prevent inadvertent use.

No. 82
COUNTRY: FRANCE
CNS-REF.-ART.: Article 14
PAGE OF REPORT: 11-12
CHAPTER OF NAT. REPORT: 2.2

Could Argentina give more information about the recommendation of the SALTO mission at Atucha I regarding the implementation of a comprehensive equipment qualification program? Is only new equipment concerned or does it also concern old equipment subject to ageing?

1. The equipment qualification program is in its establishment phase, performing the following five main tasks:
    a) Basic and detail engineering of the equipment that will be replaced by qualified equipment and preparation of the technical specifications of purchase.
    b) Purchase of the new qualified equipment and components.
    c) Technical specifications for the qualification of the remaining equipment and components that are planned to qualify for any of the other methods provided (modification, analysis, testing, dedication). Task hired to CNEA.
    d) Performing of the tasks specified in point B. (modification, tests, analysis, dedication).
    e) Mounting and commissioning for all the equipment qualified in points B and D.
2. Most of the I&C equipment will be replaced by new qualified equipment, as well as some electric equipment (e.g. cables and actuators). Old equipment already aged will be also qualified as follows: cables by type testing; actuators by analysis and type testing; and penetrations and junction boxes by modification and subsequent testing. Mechanical equipment will be qualified by analysis, operating experience or replaced (in a very few cases).

No. 83
COUNTRY: FRANCE
CNS-REF.-ART.: General
PAGE OF REPORT:
CHAPTER OF NAT. REPORT: Summary

In his report, the President of the 7th review meeting had recommended that Contracting Parties consider the implementation of the good practices that where identified during the meeting. Could your country provide information on the actions carried out with regards to the implementation of those good practices in your country?

From the four good practices identified at the 7th Review Meeting, Hungary's good practice was the driven force for improving the communication process to stakeholders.

Communication to stakeholders by ARN was improved relaunching an updated ARN website at www.argentina.gob.ar/arn with an enhanced a content, tailored to 3 different users' profile –general public, regulated and students, in a more modern and accessible website for people with different abilities, and with responsive design.

No. 84
COUNTRY: SOUTH AFRICA
CNS-REF.-ART.: Article 17
PAGE OF REPORT:
CHAPTER OF NAT. REPORT: 3.17

As part of evaluating the radiological impact of NPP operation on the public and environment, does Argentina monitor the prevalence of cancer among population groups living around NPP sites (for example, by conducting relevant research studies)?

Argentina as part of the requirements do not have Ecology, and fresh water supply. Is there any particular reason as to why?

1. There are no specific studies on the prevalence of cancer in the population around the NPPs. To evaluate the environmental radiological impact, monitoring and follow-up of authorized discharges to the environment are carried out, sampling and measurements of radionuclides are carried out in the different environmental matrices. Likewise, the dose to the representative person of each site is calculated for the emitted discharges.
2. Standard AR 10.10.1 "Site evaluation for NPP" establishes specific requirements and criteria for this purpose:
   - The environmental radiological impact must be evaluated considering all the operating states and accident conditions, including those cases that may lead to emergency measures.
   - The surrounding geographical area should be evaluated considering the foreseeable present and future characteristics and the distribution of the population, including the present and future uses of land and water, and any other characteristic that may affect the possible consequences of radioactive emissions to the public and the environment.
   - Land and water uses of the site area: land and water uses should be characterized to assess the effects of the nuclear power reactor on the site area and to prepare emergency plans. The evaluation should include land and water bodies that can be used by the public or can serve as a habitat for organisms present in the food chain.
   - Possible effects on the public should be evaluated due to the dispersion of radioactive materials, both in surface waters and groundwater, using the data and information collected.
   - Environmental radioactivity: before commissioning a nuclear power reactor, the environmental radioactivity of the atmosphere, hydrosphere, lithosphere and biota in the site area must be determined, in order to assess the effects of the operation of the nuclear power reactor. The data obtained will constitute the environmental radiological baseline and should be collected periodically for a period of at least one year, before commissioning.

No. 85
COUNTRY: SOUTH AFRICA
CNS-REF.-ART.: Article 6
PAGE OF REPORT:
CHAPTER OF NAT. REPORT: 3.6

Was the concept of extended shutdown considered prior to issuing approval for LTO phase A?

CNA I reached its end of life in 2018, current operation is stated as under LTO Phase A up until the first of 5 equivalent full power years or ten calendar years. Prior to entering LTO phase B the Global Assessment and Conceptual Implementation plan would be submitted to the Regulator in 2020. Is it expected that the projects that is in this plan be completed prior to entering or receiving approval for phase B, i.e. within 4 years if 2024 is the end date for phase A?

The concept of extended shutdown wasn´t considered.

The submission that ARN has to receive in 2020 contains the implementation plan, where a schedule for the corrective and improvement measures has to be proposed to ARN. The regulatory expectation is that the basis for the schedule be the safety significance of such measures, so a “timely implementation” can be assured. ARN expects that the plan be completed prior to entering in phase B.

No. 86
COUNTRY: SOUTH AFRICA
CNS-REF.-ART.: Article 6
PAGE OF REPORT:
CHAPTER OF NAT. REPORT: 3.6

Does ARN play an active role in accumulating and dissemination of OE and is there an internal process for this within ARN?

ARN receives immediate and 24-hour communications of all the relevant events that occurred in the three nuclear power plants of the country, such as a 60-day evaluation report. Both communications and reports are archived on a platform for storage of information and documents. Also the reports made by this ARN on these events are archived in it. Meetings are held with the group of operational experience of the plants. The dissemination of events with their staff and safety culture is required and this is measured by a performance indicator.

The way of exchange information with other countries is through the IRS. First, an analysis is made of the events that may be useful to help prevent events in other nuclear power plants around the world. Then they are sent and loaded into the IRS database, which includes events from all the member countries of the group.

No. 87
COUNTRY: SOUTH AFRICA
CNS-REF.-ART.: Article 6
PAGE OF REPORT:
CHAPTER OF NAT. REPORT: 3.6

Has Argentina adopted the revised dose limits to the lens of the eye and the extremities? If so, is it implemented or by when will it be implemented?

Does Argentina have any regulations regarding protection of the biota? If so, what is the status of implementation? If not, is it planned to be developed and implemented?

Standard AR 10.1.1. Basic Radiological Safety Standard establishes the following dose limits for workers:

- an equivalent dose to the lens of 20 mSv per year. This value should be considered as the average in 5 consecutive years (100 mSv in 5 years), not exceeding 50 mSv in any of the individual years.
- an equivalent dose to skin or extremities of 500 mSv per year.

While the dose limits for the public are:

- an equivalent dose in the lens of 15 mSv per year
- an equivalent dose in skin or extremities of 50 mSv per year.

For students 16 to 18 years of age, who in their studies require the use of radiation sources, the equivalent dose limit to the lens is 20 mSv per year and the limit of equivalent dose in skin or limbs is 150 mSv per year.

Standard AR 10.1.1. Basic Radiological Safety Standard establishes the requirements for the protection of people and the environment against the harmful effects of ionizing radiation and for the safety of radiation sources.

Standard AR 10.10.1 “Site evaluation for nuclear power plants” establishes the following requirements related to environmental protection:

D1. General requirements

21. The environmental radiological impact must be evaluated considering all the operating states and accident conditions, including those cases that may lead to emergency measures.

D2. Requirements Associated to the effects of the nuclear power reactor on the site area

27. The site assessment should identify and analyze the direct and indirect routes through which radioactive material can reach the public and the environment, to determine the potential radiological impact of nuclear installations in the site area.

28. The location and design of the nuclear power reactor should be assessed together to ensure that the radiological risk to the public and the environment is as low as reasonable to achieve.

D5.5 Environmental radioactivity

70. Before commissioning a nuclear power reactor, the environmental radioactivity of the atmosphere, hydrosphere, lithosphere and biota in the site area must be determined, in order to assess the effects of the operation of the nuclear power reactor. The data obtained will constitute the environmental radiological baseline and should be collected periodically for a period of at least one year, before commissioning.

ARN has the CROM V8 model as a tool to perform specific estimations in biota.

No. 88
COUNTRY: SOUTH AFRICA
CNS-REF.-ART.: Article 13
PAGE OF REPORT: 84
CHAPTER OF NAT. REPORT: 3.13.2.1

CNE-LTO Project: It is mentioned in this section that in certain cases third part inspectors are delegated to audit suppliers. In which cases are the third party inspectors delegated?

Inspections of all national suppliers were performed by the LTO staff.

For the fabrication of reactor internals, pressure tubes, calandria tubes, closure sealsand feeders, and steam generators, resident inspectors were assigned in the factory during the whole fabrication process, and periodic audits and surveillances were performed by the LTO QA group.

In the case of international suppliers third party controls were performed.

No. 89
COUNTRY: SOUTH AFRICA
CNS-REF.-ART.: Article 14
PAGE OF REPORT:
CHAPTER OF NAT. REPORT: 14

What were the major issues experienced during start-up of the CNE plant under the new power uprated conditions?

What has been the impact of the power uprate on in-service inspection programme?

The major issues experienced during CNE Start-up Process were:

- Shutdown system #1 trip when a High Pressure of the Primary Heat Transport test procedure was being carried out.
- Turbine trip after Reactor setback test.
- Turbine manual trip after grid operator required reactive power decrease.
- Shutdown System #2 trip by High Neutronic Power.
- Shutdown System #1 trip due to Steam Generators Low Level during the performance test instruments installation and enabling.
- Loss of power in Metal Clad #3O bus bar Class III.
- Turbine trip due to Steam Generator #3 High Level.
- Complete loss of offsite power after argentine interconnection system breakdown.
- Shutdown system #1 actuation due to low level on Steam generator #2.
- Unnoticed discontinuity of ""Online Gas Chromatography"" measurement data display.

Impact of the power uprate on in-service inspection programme

The CNE's In-Service Inspection (ISI) Programme incorporated design changes performed in the BOP (Balance Of Plant), in which Feed Water pre-heating systems were modified. Regarding the design changes in the NSP and BOP, equipment were replaced with the same characteristics as the ones being replaced. These replacements were considered in the Embalse ISI Programme so they fit within the CSA 285.4 Standard regarding inspection periods and baseline inspections.

No. 90
COUNTRY: SOUTH AFRICA
CNS-REF.-ART.: Article 18
PAGE OF REPORT: 148
CHAPTER OF NAT. REPORT: 3.18.1

Given that CAREM is an innovative design are there any design specific requirements applicable to it, in addition to those in SSR-2/1 Rev. 1, and what are they.

The design of the CAREM 25 Reactor is prior to the issuance of the SSR-2/1 Rev. 1, however, it is harmonized with the requirements of this guide and the safety requirements of coming from the post Fukushima accident lessons were included in the design even before the accident.

In the framework of the Licensing scheme, considering CAREM 25 as a prototype of NPP, ARN granted the authorization for construction with "license conditions" which was a set of regulatory conditions that reinforcing authorization, as a result of the assessment of a safety demonstration based on comprehensive deterministic and probabilistic safety analysis.

No. 91
COUNTRY: SOUTH AFRICA
CNS-REF.-ART.: Article 18
PAGE OF REPORT: 148
CHAPTER OF NAT. REPORT: 3.18.2

Did ARN develop any special regulatory requirements for a NPP that uses passive safety systems?

No regulatory standard has been developed that contains requirements for a NPP that uses passive safety systems. Regarding to Licensing of CAREM 25, as mentioned in section 2.4, one of the areas with special attention was the validation and verification of computer codes used for the RE for the demonstration of Safety.

ARN does not have specific standards for the system codes used for safety analysis / demonstration (TH, neutronics, fuel behavior, etc.) but the most commonly used codes have been already validated and verified within respective correlations range (RELAP5, TRACE5, CATHENA, CATHARE, MARS, ATHLET, and others).

However, due to special design features of CAREM reactor, ARN required a comprehensive analysis of each conditions and physical phenomena that could occur during PIEs transients/accidents and checking codes capability for capturing and representing them. This procedure holds mainly for thermal-hydraulics and neutronics codes (cell and core level) and some specifics codes have been developed for engineering / design, e.g. steady-state conditions, DNB, instabilities maps.

To develop this task, ARN provided guidance (based on CNSC RG G-149) on how to develop models / codes, its documentations and the quality assurance process required.

No. 92
COUNTRY: SLOVAKIA
CNS-REF.-ART.: Article 16
PAGE OF REPORT: 125
CHAPTER OF NAT. REPORT: 3.16.4

The report says that in case of Green Alert, Argentina intends to distribute stable iodine tablets to the population living in the area of UPZ. As it is likely, that such distribution could take some time, are there any future plans to distribute iodine tablets in advance before the accident?

Currently, stable iodine tablets are stored in the NPP and at strategic points in the cities involved. Argentina doesn´t rule out the possibility of implementing a distribution strategy differently in the future.

No. 93
COUNTRY: SLOVAKIA
CNS-REF.-ART.: General
PAGE OF REPORT: 12
CHAPTER OF NAT. REPORT: 2.3 CNA II

ANSYS/CFD code probably does not cover secondary contamination of airflow called radioactive aerosol evaporation. How the contingency for active radionuclide evaporation is calculated (i.e. OMEGA Software recommended by IAEA)?

Exactly as you mention, the ANSYS/CFD code does not cover the contamination assessment.

The ANSYS/CFD calculations are used to assess whether or not the early containment bypass (corium melting the safety injection suction lines) can be avoided. In case it could, then an early large release from the containment to the environment can be precluded. Our goal is to avoid the containment bypass.

In the past for Atucha II PSA-L2 we employed MELCOR code for calculating radionuclide behavior inside the containment.

No. 94
COUNTRY: SLOVAKIA
CNS-REF.-ART.: General
PAGE OF REPORT: 13
CHAPTER OF NAT. REPORT: 2.5

...an agreement between NA-SA and Almafuerte Firefighters Station located more than 15 km from the Embalse NPP....

The description does not provide an information about the reaction time to cover emergency, we recommend to use reaction time rather, i.e. 15 minutes.

In case of internal emergency, CNE immediately calls Embalse firefighters, which take between 14 and 18 minutes to arrive, according to drills and previous experience. It is also established that, if convened, the emergency response organization must meet at the Internal Emergency Control Centre within 60 minutes of the calling.

In case of requiring external assistance for an event within the site, CNE immediately calls the firefighters of the city of Embalse (they are 7 km from the NPP), which take between 14 and 18 minutes to arrive, according to the drills and previous experience.

On the other hand, if a general emergency is declared in the NPP, the external response organizations are notified immediately to assist the COEM that it would be temporarily located in the fire department of the city of Almafuerte (approximately 18 km from NPP).

No. 95
COUNTRY: SLOVAKIA
CNS-REF.-ART.: Article 6
PAGE OF REPORT: 23
CHAPTER OF NAT. REPORT: 3.6.5.1

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 extension for CNE I from 1974 up to the year 2049 or 2054 seems to be very long. So long extension may lead to significant material degradation, which could be not executed by mitigation measures. What are the main arguments for such a long term operation?

Section 3.6.5.1 refers to CNE Life Extension and the related activities in order to safe operate the plant for 25-30 additional years depending on the load factor of the plant.

We understand that the question was posted for CNA I. In this case, the basis for definition of intended continued operation time was the demonstration (with safety margin) of the structural integrity of the reactor pressure vessel up to the end of the long term operation. It was also demonstrated for such period of time, the structural integrity of the metallic containment. So, the critical components were demonstrated to be fit for that additional period of service.

No. 96
COUNTRY: SLOVAKIA
CNS-REF.-ART.: Article 9
PAGE OF REPORT: 61
CHAPTER OF NAT. REPORT: 3.9.3.5

Does the operator intend to construct an “information centre” near by the NPP?

An “information centre” is a useful tool that has proven successful in other projects with similar characteristics. The company has considered the implementation of an information centre, however due to the embryonic state of the new builds, we have not yet advanced in the discussion of an “information centre” for what could be Atucha III.

No. 97
COUNTRY: SLOVAKIA
CNS-REF.-ART.: Article 11
PAGE OF REPORT: 68
CHAPTER OF NAT. REPORT: 3.11.1

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.

The financial resources cover also the costs for decommissioning of NPP´s? There is no information about covering these costs from the electricity fees.

Decommissioning: In accordance with the provisions of Article 2, subsection e) of Law No. 24.804, it is the responsibility of the National Atomic Energy Commission to determine the manner of decommissioning of nuclear power plants; the scheme presented has not yet been approved.

In accordance with the provisions of PEN Decree No. 1.390/98 regulating Law No. 24.804 on nuclear activity, the fund with the necessary resources to face the decommissioning from each nuclear power plant would be created with the contributions of the company that became an operator of nuclear power plants to be privatized. Law No. 26.784, in its article 61, repeals article 34 of Law No. 24.804, so NA-SA is no longer subject to privatization. For this reason and in accordance with the provisions of article 37 of Law No. 24.065, which states that "the generation and transportation companies of total or majority ownership of the national State will have the right to recover only their total operating and maintenance costs that allow to maintain the quality, continuity and safety of the service…" the responsibility of financing the decommissioning of nuclear power plants is not assumed to date by NA-SA.

Similarly, it is noted that to date the sixth National Report 2017 of the "Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management" has been issued. In its point F.6.5 it states that, the financing for the constitution of said fund must be assumed by the national State with own funds.

No. 98
COUNTRY: SLOVAKIA
CNS-REF.-ART.: Article 19
PAGE OF REPORT: 177
CHAPTER OF NAT. REPORT: 3.19.9.2

Wet interim storage. Afterwards, treatment and conditioning for final disposal.

A dry interim storage is not considered due to lower operational costs, as it is stated in the Chapter 3.19.9.3.1?

The statement "Wet interim storage. Afterwards, treatment and conditioning for final disposal" is for research reactors. The strategy was a decision not connected with the spent fuel treatment and storage at CNA I NPP, as described in section 3.19.9.3.1.

No. 99
COUNTRY: SLOVAKIA
CNS-REF.-ART.: Article 19
PAGE OF REPORT: 178
CHAPTER OF NAT. REPORT: 3.19.9.3.1

Is there in the Wet interim storage an emergency wet pool, for example in case of maintenance?

There is no emergency wet pool in the Wet interim storage. But we can isolate the different spent fuel pools to perform maintenance on the working pool.

No. 100
COUNTRY: GERMANY
CNS-REF.-ART.: General
PAGE OF REPORT: 7
CHAPTER OF NAT. REPORT:

According to the National Report the assessment of the habitability of the MCR and SCR is an on-going activity. Does this activity comprise all three NPPs in operation (CNA I/CNA II/ CNE)? What are the main aspects to be assessed?

The activities comprise CNA I and CNA II NPPs.

The main activities that were performed are the following:

- A verification of the design of the main and auxiliary control rooms was carried out during 2016.
- 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 current 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.

No. 101
COUNTRY: GERMANY
CNS-REF.-ART.: General
PAGE OF REPORT: 14
CHAPTER OF NAT. REPORT: 2.6

In the National Report it is stated that CNE is currently operated under the commissioning licence and that the mandatory documents to apply for the operating licence are under preparation. Could Argentina please share the status of the commissioning program of CNE? Has Phase C "increasing the power of the reactor until reaching 100% FP, executing all the tests defined" successfully finished? Is commercial production of electric energy permitted under the commissioning licence? What are the regulatory limitations to avoid a long period of full power operation without a valid operating licence?

The licensee had already executed all the tests corresponding to phase C of the commissioning.

During the commissioning phase, the licensee conducted tests of various types to verify the operation of the plant according to the design, taking into account all the design changes that were implemented. In addition, ARN requested the licensee to incorporate additional tests to the program. For example:

- Power ramp at 30%FP, 50%FP and 75%FP to verify the control capacity of the plant to reach the new set power.
- Manual Set-back from 50%FP to 20%FP, 80%FP to 50%FP and 100%FP to 80%FP to verify the control capacity of the plant to reduce the power.
- Manual turbine trip at 50%FP, 80%FP and 100%FP to verify the automatic power reduction functions (step-back) and the capacity of the bypass valves.
- Main Steam Safety Valves Capacity test at 50%FP.
- Partial loss of feedwater to a simple steam generator.
- Shut Down System Nº1 manual trip to verify the dynamic response of the plant and power recovery.
- Load rejection from 500 kV line at 50%FP.

All tests were successfully carried out by licensee and were oversee by the inspectors of the ARN. All the tests at 100% FP and phase C of the commissioning was successfully completed.

In order to authorize the licensee to perform all commissioning tests up to 2064 MW (100% FP) of nominal thermal power corresponding to phases B and C of the commissioning program, the ARN issued an amendment to the Operation License of the Embalse Nuclear Power Plant. The validity of this license would be as long as phases B and C tests were carried out. It was expressly established in this license that, once the last phase C test has been executed and approved by ARN, the licensee must request, within a maximum period of 30 days, the issuance of the new operating license so that the installation can continue operating.

Thus, the regulatory limitation to avoid long periods of operation at full power without a valid Operating License was precisely established in the amendment to the license that allowed the licensee to perform phase B and C tests of commissioning.

No. 102
COUNTRY: GERMANY
CNS-REF.-ART.: Article 6
PAGE OF REPORT: 21
CHAPTER OF NAT. REPORT: 3.6.4.1.2

According to the National Report at CNA I a temporary emergency control room was installed. Could Argentina please explain why this emergency control room is only temporary? In our understanding SSR 2/1 requires a permanent supplementary control room (see Req. 66).

The actual emergency control room in CNA I is defined as temporary because it is not the definitive one which will be constructed, commissioned and put into operation for entering into the phase B of the LTO project.

The original design of CNA I didn't contemplate the existence of an emergency control room. After Fukushima accident and later, as a result from the Periodic Safety Review done in 2014, became clear the necessity to solve this non-compliance in the light of modern safety requirements.

Due to a feasibility study done by the plant, ARN decided to require the installation of an emergency control room divided in a two-step process.

By the first step and in order to enter into the phase A of the LTO project, ARN required to install this temporary emergency control room in the Secondary Heat Sink building from where the plant can be shut it down and keep it in safe state. Also, instruction for operation and monitoring the plant from this temporary emergency control room were developed.

By the second step and in order to enter into the phase B of the LTO, a "definitive" emergency control room has to be installed. The design basis of this emergency control room has to be equivalent to the corresponding at CNA II. ARN is waiting for the implementation plan which will be submitted for approval next march 2020.

No. 103
COUNTRY: GERMANY
CNS-REF.-ART.: Article 6
PAGE OF REPORT: 21
CHAPTER OF NAT. REPORT:

It remains unclear, if the temporary emergency control room serves CNA I as well as CNA II. Are there two emergency control rooms (one for CNA I and one for CNA II) or can both reactors be controlled from a single emergency control room?

CNA II has operative its own emergency control room. Instead, CNA I has operative its temporary emergency control room and has the plan to construct- commission and put in normal operation the permanent emergency control room, in the frame of the LTO project.

So, each plant can be controlled from the corresponding emergency control room.

No. 104
COUNTRY: GERMANY
CNS-REF.-ART.: Article 6
PAGE OF REPORT: 23
CHAPTER OF NAT. REPORT: 3.6.4.3.3

Could Argentina please describe the principle of the level measurement of the spent fuel pool utilizing a compressed air bubbler?

The principle of operation is quite simple. In the Bubbler level gauge, a bubbler tube is used to measure and indicate pool level. Air is forced through the bubbler tube and the bubbles come out on the bottom of the liquid level. The greater the height of the liquid in the pool, the more pressure it needs to push the air out.

In this way, the back pressure from the bubbles is measured and converted into level value.

No. 105
COUNTRY: GERMANY
CNS-REF.-ART.: Article 6
PAGE OF REPORT: 29
CHAPTER OF NAT. REPORT: 3.6.5.2

In the National Report the main activities of CNA I LTO stage "B" are listed. Based on the National Report it seems that some of the activities have already been completed and some are ongoing. Could Argentina please clarify, which activities have already been implemented, and comment on the time schedule for completion of the remaining activities?

The activities that were already implemented in order to enter into phase A, are:

- Condition assessment of all in scope systems, structures and components (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. Updating the P-T curve;
- Development of equipment qualification master list (environmental, seismic and electromagnetic immunity) and the program for further qualification;
- Completion of implementation of corrective measures (as far as practicable) from the review of safety factors belonging to the last PSR (2014) according to the IAEA SSG-25;
- Plant implementation of relevant 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;
- Implementation of temporary emergency control room.

Conceptual Implementation plan for the remaining activities will be submitted to ARN in 2020. It is expected that the projects that are in this plan be completed prior to entering into phase B, which will be in 2023 or within 4 years (as per the operating licence), depending when is the end date for phase A (2023 or 2024).

No. 106
COUNTRY: GERMANY
CNS-REF.-ART.: Article 7
PAGE OF REPORT: 37
CHAPTER OF NAT. REPORT: 3.7.3.2.1

It is stated in the National Report that in 2010 ARN introduced the "ad hoc" licensing scheme for the innovative CAREM reactor. Later ARN revised the licensing scheme. Could Argentina please share its experience why the "ad hoc" licensing scheme will not be followed?

The evolution of the project and the experience gained in other projects (OL of CNA II, LTO license of CNE, pre-licensing of an updated CANDU and of the HPR 1000) leads to an up-date of the licensing scheme of CAREM 25.

The revised licensing scheme fits completely in the licensing procedures foreseen for new NPPs in terms of mandatory documents (table of contents and scope) and overall approach.

No. 107
COUNTRY: GERMANY
CNS-REF.-ART.: Article 7
PAGE OF REPORT: 37
CHAPTER OF NAT. REPORT: 3.7.3.2.1

As stated in the National Report Argentina issues commissioning licences. Do these licences include hold points in the commissioning activities? In case of, could Argentina please share the defined hold points for Embalse as well as the foreseen hold point for commissioning of the CAREM reactor?

The commissioning license authorized the licensee to perform all commissioning tests up to 2064 MW (100%) of nominal thermal power to execute phases B and C as described in the Licensing Bases Document (LBD) for Embalse Life Extension Project. In this LBD, which was part of the mandatory documentation established in this commissioning license, all tasks of regulatory interest, including commissioning, were established with licensing milestones and hold points.

In summary, the following milestones and their corresponding prerequisites were established.

Milestone 1: Hydrostatic Test of Primary Heat Transport System (PHTS).

Prerequisite 1: Containment system available and verification of proper conservation of systems.

Milestone 2: Start fueling the reactor.

Prerequisite 2: Successful Hydrostatic Test of PHTS and notice ARN 60 days before.

Milestone 3: Guaranteed Shutdown State removal and power increase up 5%FP.

Prerequisite 3: Containment leak test, conformation of the ad hoc Committee for commissioning and mandatory documentation defined in the LBD approved (Phase B).

Milestone 4: Authorization of tests at different power levels (5%FP, 50%FP, 80%FP and 100%FP) (Phase C).

Prerequisite 4: Successful tests of the previous power level and evaluation of the corresponding report issued by the ad hoc Committee.

Milestone 5: Application for the Operation License.

Prerequisite 5: All mandatory documentation analyzed and approved by ARN including the final report of the ad hoc Committee.

Regarding CAREM:

The Commissioning Program (CP - included in the IS as chapter 14) defines a set of tests that will be carried out in the installation in order to demonstrate that the safety objectives of the SSC design are met and that the installation will work with safety conditions in both normal and abnormal operation. Therefore, all aspects of the planned operation for the Installation Operation stage must be consolidated to include them in the CP and test them during the Commissioning stage. Due to the design characteristics of the CAREM 25 prototype, the CP will require special treatment by the Responsible Entity (RE) to deal with states or situations outside the nominal parameters or with functional conditions not yet characterized. This includes, for example, tests at low and intermediate powers with a natural circulation greater than expected by calculation. This requires an early definition of possible values for the Operating Limits and Conditions (OLC, chapter 16 of the Safety Report) and a proposal of the detention points during the tests. These RE proposals must be approved and authorized by the ARN.

No. 108
COUNTRY: GERMANY
CNS-REF.-ART.: Article 7
PAGE OF REPORT: 38
CHAPTER OF NAT. REPORT: 3.7.3.2.2

It is stated in the National Report that for CNA II the operating licence (issued in May 2016) is only valid for 5 years. In response to question No. 30 from CNS 2017 Argentina answered that the validity of the operating licence will be increased after the planned outages in years 2017 and 2018. Has the validity of the operating licence for CNA II increased to an interval of 10 years? Does it require a PSR for a license renewal? When will the first PSR for CNA II be performed?

No, the validity of the operating licence for CNA II has not increased to an interval of 10 years. According to an Act, signed by Boards of Directors of both Operator and Regulatory Body, considering the need of capitalizing the experience of having completed a “delayed project” and applying the principle of continuous improvement of nuclear and radiological safety, was decided to stablish a 5 years period for the first operating licence for CNA II.

In this Act were stablished the requirements for renewal the operating licence, including among others, the results of tasks done during the planned outages in years 2017 and 2018. Also there were requirements in relations with: evaluation programme of safety systems, extension of probabilistic safety analysis and technical specifications. The compliance of all these items will be the base for renewal the operating licence for another 5 years period.

However, if Operator presents the PSR by this time, the operating licence could be renewed for 10 years. If not, the Regulatory Body will require the Operator to present the first PSR at least for petition of licence renewal after 10 years operation.

No. 109
COUNTRY: GERMANY
CNS-REF.-ART.: Article 7
PAGE OF REPORT: 39
CHAPTER OF NAT. REPORT: 3.7.3.3

It is stated in the National Report that ARN performs regulatory audits to analyse 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.

Does ARN also assess the safety culture at a specific NPP during such audits?

How are observations regarding safety culture obtained and taken into consideration from planned inspections and special/reactive inspections?

Does ARN apply performance indicators for its assessment?

1) Currently ARN does not assess organizational safety culture (SC) as whole during regulatory audits to NPPs.
   - "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." (NR, 3.8.3.1.2). This standard does not have regulatory requirements on safety culture, as stated by standards coherent with IAEA GSR Part 2.
   - "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." (NR, Question No. 43, Annex II).

   However, ARN assess some SC topics during regulatory audits to NPPs.
   - "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… 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.. " (NR, 2.9)
   - Previous regulatory audits covered some PRACS topics (i.e. regulatory audit to CNA I operating experience programme).
2) See 3.10.2.2 Safety culture and its developments (i.e. evaluation of the SC for renewal of personnel specific authorizations, evaluation of the SC attitudes during inspections, etc.).
3) Currently:
   - ARN does not apply safety performance indicators (SPI) to assess NPP safety culture;
   - ARN is developing a methodology to assess SPI related to NPP organization (safety attitude, internal control and compliance).

No. 110
COUNTRY: GERMANY
CNS-REF.-ART.: Article 8
PAGE OF REPORT: 54
CHAPTER OF NAT. REPORT:

According to the National Report ARN staff was reduced by approximately 15%. Could ARN please explain in more detail the reasons for this reduction? How does this reduction affect ARNs performance reflecting the assessment of one reactor in commissioning, one reactor

under construction and an expected review of the design assessment for a construction licence? How will ARN manage the workload with reduced human resources and at the same time adequately ensure its regulatory functions?

Nowadays, ARN is subject to the restrictions in force for institutions of the National Public Administration to hire or contract personnel. Since 2017 there has been a policy of disengaging personnel with age over the retirement limit, and offering of voluntary retirements to personnel under that age. ARN coped with the situation transferring personnel with competences to critical activities from sectors that carry out activities of lower priorities.

No. 111
COUNTRY: GERMANY
CNS-REF.-ART.: Article 14
PAGE OF REPORT: 105
CHAPTER OF NAT. REPORT: 3.14.3.1.4

It is stated in the National Report that more than 40 sequences were analysed by deterministic safety assessments for the CAREM reactor. SSR 2/1 requires that the design is capable of coping with accidents more severe than design basis accidents. Despite ATWS, what kind of accident sequences have been analysed as DEC without core melt? How are such accident sequences derived?

The phenomenology involved in accidents with core fusion (severe accidents) differs radically from those where there is no core fusion. Therefore accidents with core fusion must be treated at a specific Defense in Depth (DiD) level. Design elements that aim to prevent core fusion conditions and that are taken into account in the demonstration of safety, do not belong to the same level of DiD as the design elements whose objective is to control severe accidents that were not prevented. That is why the objective of Level 4 Defense in Depth is the control of accidents with core fusion -postulated- to limit off-site release, as far as reasonably practicable.

No. 112
COUNTRY: GERMANY
CNS-REF.-ART.: Article 18
PAGE OF REPORT: 148 + 151
CHAPTER OF NAT. REPORT: 3.18.3.1.1

It is stated in the National Report that in preparation of LTO for CNA I, a benchmark against the most recent revision of KTA standards (KTA 3206, KTA 3501 and KTA 3904) was performed. Could Argentina please share the identified safety improvements related to 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"?

As part of the LTO project it is planned to replace the Reactor Protection System (RPS). The design and construction of the new RPS will be based on current normative (IEEE-1E or/and KTA 3501).

Regarding KTA 3904, the main discrepancy is the lack of an emergency control room (ECR), which is planned to be constructed as part of the LTO Project. The ECR will be constructed according to the regulatory requirements and following the current international normative. Another aspect is the need to improve the ventilation system of the MCR to ensure its habitability in the presence of smoke or harmful gases outside. The "Habitability of the MCR" Project is currently in course.

The gap involving automatic fire extinction system was resolved last year.

No. 113
COUNTRY: GERMANY
CNS-REF.-ART.: Article 19
PAGE OF REPORT: 160-182
CHAPTER OF NAT. REPORT: 3.19

On 16 June 2019 Argentina was faced with power failures in large parts of the country, having also an impact on all three Argentinian NPPs in operation. How long were CNA I, CNA II and CNE facing a loss of offsite power? Does Argentina classify this loss of offsite power as an AOO or DBA? What counter measures are in place to cope with such a situation (house load operation or emergency power supply)?

The loss of offsite power is treated as an AOO. A plant facing this event can go to a safe shutdown condition or to a reduction of power to self-consumption disconnected from the grid. Emergency Power Supply is available using diesel generators.

- Atucha I was already in shutdown condition and Emergency Power started successfully.
- Atucha II was at 40% of full power and continued operating in self-consumption.
- Embalse shutdown when the offsite power was lost and diesel generators started successfully. The loss of offsite power for Embalse lasted more than 9 h and the plant was in shutdown condition for 40 h.

This event in 2019 was extraordinary because it persisted for about 13 h within almost all the country, but no consequences were reported in any NPP.

It is worth mentioning that one of the items of the stress test performed in all NPPs after Fukushima was the extended Loss of Offsite Power including Station Black Out.

This event has not been included in the main report because it happened on 16 June 2019, after the closing date of March 2019.

No. 114
COUNTRY: INDIA
CNS-REF.-ART.: General
PAGE OF REPORT: 3
CHAPTER OF NAT. REPORT: 1.3

It is mentioned that CNNC is the supplier for HPR 1000 PWR and NA-SA holds the Design Authority role. ARN and NA-SA have signed a MoU on regulatory requirement.

Does ARN plan to undertake a detailed review and assessment of the overseas HPR 1000 design for its licensing in Argentina?

Yes, according to the licensing process of a NPP in Argentina a detailed review and assessment of the HPR 1000 design has to be performed for granting the operating licence.

MOU must be understood as a pre-licensing activity and doesn’t diminish the regulatory involvement during the licensing process. According to Argentinean standard, it is necessary to review and further approve the Preliminary Safety Analysis Report and all the topical related documentation as a condition for granting the operating licence.

It is important to stress that ARN is member of Multinational Design Evaluation Program HPR1000 working group (MDEP-HPR1000).

No. 115
COUNTRY: INDIA
CNS-REF.-ART.: Article 10
PAGE OF REPORT: 65
CHAPTER OF NAT. REPORT: 3.10.2.2

It is mentioned that the Licensee has developed PRACS to reinforce nuclear safety culture and PRACS performance is measured through Surveys.

Could Argentina share:

1) **The process of PRACS?**
2) **Some examples of recommendations of PRACS committee.**

1) The PRACS process consists of selecting specific topics in the organization (fire protection, emergency preparedness, equipment reliability, human performance, operational decision making, indicator management, among others), through the definition of persons responsible (coordinators), in the different NA-SA sites, who lead these issues, which must implement a specific methodology (meetings and deliverables) to carry out transversal objectives to the organization, through the fulfilment of specific actions. The status of the issues is monitored in the framework of a PRACS committee (at a plant site and headquarters level) which is led by the corresponding site manager (in the case of headquarters, the committee leader is the General Manager).
2) Some examples of recommendations arising from the PRACS committees are: process and procedures unification between the 2 NA-SA power plants sites (eg, error prevention techniques, corrective actions, operational decision making, emergency plan, self-assessments, indicators); development of plans for joint communication between the 2 NA-SA power plants sites (eg industrial safety topics).

No. 116
COUNTRY: INDIA
CNS-REF.-ART.: Article 13
PAGE OF REPORT: 83
CHAPTER OF NAT. REPORT: 3.13.2

It is mentioned in the report that NA-SA has a Quality Assurance program to ensure that SSCs meet the necessary standards. ARN also audits the Quality Programs of the NPPs.

Can Argentina clarify whether ARN also have any plans to monitor in future at Suppliers' premises as part of Supply Chain Management? (for example: in the case of HPR 1000).

As part of the regulatory involvement during the licensing process ARN performs audits at manufacturer shops with the purpose to assess the application of the Quality Assurance program in their activities. ARN gives special attention to the fulfilment of the SSC´s engineering requirements (robustness, functional capacity and reliability, all supporting the safety demonstration) during the whole life including the manufacturer stage.

At certain cases when the licensee decides to nominate a certified third party in the project, ARN adapt its involvement during product realization by oversighting the results of the third party inspections.

No. 117
COUNTRY: INDIA
CNS-REF.-ART.: Article 14
PAGE OF REPORT: 92
CHAPTER OF NAT. REPORT: 3.14.2.2

It is mentioned that during the retubing campaign (2016 to 2018) all the pressure tubes and calandria tubes were replaced by new ones with improved design features.

Can Argentina share information on these improved design features?

The design changes were proposed by CANDU Energy, depending on the operational experience, to comply with the design requirements of the fuel channels.

- Pressure tubes: The new design of this component includes changes in geometry, material chemistry and the tests required during manufacturing.

  The geometric change proposed by the designer was to increase the thickness of the tube from 0.165 "to 0.169", maintaining the internal diameter and with a slight increase in the external diameter. As is known, one of the effects of irradiation-induced creep is the dimensional change of the pressure tubes without a change in volume, which leads to a decrease in the thickness. The modification proposed by the designer aims to ensure thickness at the end of life, which implies an increase in tensions, to be sufficient to withstand the tensions imposed by the different states of load including accidental ones.

  Based on the operational experience of CANDU 6 reactors, the designer proposed changes in the chemical composition of the pressure tube aimed at decreasing its probability of failure. These changes correspond to reducing 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. These changes allow:

  - Decrease the amount of initial hydrogen, which influences the phenomenon of DHC and blistering.
  - Improve the strength of the material.
  - Improve corrosion performance.
  - Improve fracture toughness.

- Calandria tubes: The most important modification to the calandria tubes corresponds to the implementation of a surface treatment for the central outer surface of the tube called "Glass peening". This treatment generates a rougher surface of the material, resulting in a better heat transfer to the moderator in the event of an accident, mainly because the vapor bubbles that form on the surface would be smaller and more easily released.

  In relation to geometric changes, the modifications were:

  - Increase of eccentricity tolerance in the flared section from 0.015 "to 0.020".
  - Extension of the perpendicularity tolerance at the ends of 0.02 "to 0.03".
  - Maximum ovality decrease from 0.04 ”to 0.02” in the central part of the tube.

These modifications would favor non-interaction between the calandria tube and the pressure tube.

No. 118
COUNTRY: INDIA
CNS-REF.-ART.: Article 17
PAGE OF REPORT: 139-141
CHAPTER OF NAT. REPORT: 3.17.2.3.2.1.2

1) **It is mentioned in the report that in 2016 CNA estimates for extreme rise of Río de La Plata River a 1,000 years recurrence was taken. For minimum water level, a minimum Paraná River flow with a 100 years recurrence was considered.**
   Can Argentina clarify whether these recurrence periods are in line with Argentine codes?
2) **It is mentioned in the report the PMH level is 8.45 m and the plant main buildings are on 23m high plateau.**
   Can Argentina clarify whether during PMH Level flooding incidence, will access to plant by personnel be hampered?

1) There is no specific argentine regulation establishing recurrence periods that should be used for risk evaluation. The rule “AR 10.10.1. “EVALUATION OF NUCLEAR POWER PLANTS SITING - Revision 0”, released by ARN (National Nuclear Regulator) establishes some general criteria in D4.2 Meteorological Phenomena:

   41. The extreme values of meteorological variables and phenomena must be evaluated, as well as meteorological and climatological features of the reactor siting. Accordingly, wind, rainfalls, snow, temperature, watercourse and tide levels parameters shall be documented for a certain period of time in order to evaluate possible extreme values.

44. The siting zone must be assessed to establish the possibility of floods or downspouts (…). If such possibility exists the relevant data must be collected and evaluated, including historical, meteorological and hydrological data.

2) The pump house is only located at river level, all the other main buildings are located on the top of the Paraná river cliff. The access to plant by personnel cannot be hampered because the entries to the plant's main buildings are located on the opposite side of Paraná River, on a 23 m high plateau.

No. 119
COUNTRY: INDIA
CNS-REF.-ART.: Article 19
PAGE OF REPORT: 163
CHAPTER OF NAT. REPORT: Section 3.19.4

It is mentioned that the NPPs have in force surveillance program for long lived components, as the reactor pressure vessel and its internals.

Can Argentina elaborate its regulatory requirements with respect to residual life assessment of the long lived components (such Calandria, Endshield etc.)?

Regulatory requirements for life assessment of long lived SSCs are oriented to demonstrate the capability to perform the intended functions during the entire period of life extension. In order to reach this safety objective ARN identified a set of regulatory expectations, which were requested to the plant. These are the following:

- In LA all components are important, so no screening was required.
- LA must include a review of applicable codes, standards and a thorough evaluation of:
  - Design margins,
  - Stress and seismic analysis reports,
  - Manufacturing processes, quality control, material/manufacturing defects.
- Maintenance, surveillance and inspection history, must comprehensively considered.
- Review of chemistry including chemistry specifications, and an analysis of chemistry excursions and their impact on materials.
- Operating/event history. Data gathering and review should include, but is not limited to:
  - failure rate history of components/subcomponents
  - component/subcomponent replacement history (including any actions arising from EQ requirements)
  - failure mechanisms observed to date
  - degradation sites
  - age related failure modes
  - service conditions (environmental, loading and power conditions resulting from normal operating requirements)
  - vibration data
  - operational transients
  - engineer's/operator's logs
  - conditions that prevail during testing, shutdowns and storage
  - unplanned events
  - thinning rates (where applicable -in order to evaluate design margins and estimate the remaining life by comparing thickness measurements against minimum thickness required)
- A qualitative fatigue assessment may be done as part of a LA such that a much more in depth analysis of operational transients may be required.
- Ageing evaluation (Ageing Management Review).
- Health prognosis.

Each piece of information presented in the aging evaluation has to be used to provide a prognosis for continued operation of the component and to provide the technical basis for recommendations.

The assigned health prognosis may be excellent/good/fair/poor for continued operation of the SSC to reach design life (life attainment) and for life extension beyond refurbishment.

The definitions of these categories are the following:

Excellent: The SCC is likely to operate beyond life attainment. No aging degradation was detected. Maintenance, health monitoring and other aging management activities are generally considered to be adequate.

Good: The SCC is likely to operate satisfactorily over life attainment with no significant aging degradation evident. That is, the SCC is generally being well maintained and does not require significant investment to reach life attainment provided that current aging management practices are continued.

Fair: The SCC is currently operating satisfactorily, but is obsolete or showing signs of aging, and may require additional remedial action for life attainment.
Improvements in aging management methods are identified to mitigate further aging degradation.

Poor: It is unlikely that the SCC will operate satisfactorily for life attainment without taking remedial action (e.g., replacement/refurbishment) within a short timeframe. Generally the SCC is exhibiting obvious signs of significant aging and/or is not in compliance with current codes/standards.

No. 120
COUNTRY: UNITED STATES OF AMERICA
CNS-REF.-ART.: General
PAGE OF REPORT: 12
CHAPTER OF NAT. REPORT: 2.4

Challenge #4 relates to licensing activities on CAREM 25, a small modular reactor prototype. Is the ARN planning to join the SMR Regulator's Forum to gather insights and best practices from the international community?

The ARN recognizes that participation in the SMR-RF sharing information regarding the licensing of the CAREM reactor would be beneficial in several aspects, but given the appropriate timing evaluation and the limitations of resources, ARN decided to postpone this participation to another stage of the licensing process. It is expected to open CAREM licensing experience when assembly and construction activities have been completed, preliminary tests are underway, and the ARN is analyzing the documentation to issue the Commissioning License. This opening will be considered considering the appropriate areas and mechanisms to integrate potential users and providers of SMRs.

No. 121
COUNTRY: UNITED STATES OF AMERICA
CNS-REF.-ART.: Article 18
PAGE OF REPORT: 158-159 and 184
CHAPTER OF NAT. REPORT: 3.18.3.5 and 4.4

Argentina has expressed interest in constructing a 4th NPP. The design selected is a HPR-100 of Chinese design. The contract between NA-SA and CNNC (China National Nuclear Corporation) has not been signed yet.

1) **What is the timeline for signing this contract?**
2) **Clarify, when is the new plan expected to become operational?**
3) **Does the cooperation arrangement between ARN and NNSA-China include cooperation for new reactors and construction activities?**
4) **Is ARN expanding its regulatory oversight program to include construction activities? If yes, please clarify to what extent.**

1) Although the negotiation of the construction contract for an HPR-1000 in Argentina is well advanced, there remain a number of substantive issues that still have to be agree among the parties before the contract can be approved and signed. The timeline for signing the contract is not yet clearly established due to the unresolved substantive issues.

2) The earliest time this plant could enter commercial operation is 2029.

3) Yes, the cooperation agreement includes the cooperation for licensing activities of new reactors.

4) Due to the recent licensing activities for Atucha II, the current regulatory oversight includes construction activities. So far, there is no need to expand it.

No. 122
COUNTRY: UNITED STATES OF AMERICA
CNS-REF.-ART.: General
PAGE OF REPORT: 11 - 13 - 15
CHAPTER OF NAT. REPORT: 2.2 – 2.6 – 2.7

Numerous activities associated to long term operation, life extension, and refurbishments are discussed. Did ARN evaluate the findings of the recent European Topical Peer Review on aging management to leverage lessons and gather good practices? If yes, were any gaps identified that needed regulatory action in Argentina?

ARN didn’t evaluate the European Topical Peer Review on aging management.

No. 123
COUNTRY: CANADA
CNS-REF.-ART.: General
PAGE OF REPORT: 6
CHAPTER OF NAT. REPORT: 1.4.1

"The regulatory conclusion resulting from the stress tests performed by the licensee and ARN is that there is a need for some regulatory actions". What regulatory actions is ARN planning to take in regards to these findings?

The main regulatory action is the updating of the AR standards embedding the lessons learnt from Fukushima Daiichi accident and the harmonization of all standards with the latest IAEA safety requirements and guides. For more information see section 3.7.2.2.

No. 124
COUNTRY: CANADA
CNS-REF.-ART.: General
PAGE OF REPORT: 16
CHAPTER OF NAT. REPORT: 2.11

"The IRRS mission was reprogrammed due to operative reasons. Does this mean the mission was rescheduled to a different date, or that the scope of the mission was changed (or something else entirely)?

The mission was rescheduled to next 4th May, 2020.

No. 125
COUNTRY: CANADA
CNS-REF.-ART.: Article 6
PAGE OF REPORT: 30
CHAPTER OF NAT. REPORT: 3.6.6

"The concept of defence in depth of the existing NPP's remains acceptable, like the one in CNA II or was/is being upgraded like in CNE and CNA I." Is there a timeline for the upgrades at CNE and CNA I?

In the case of CNE, the timeline for upgrade the defence in depth concept was connected to the LTO project. All the measures for the safety upgrade were mostly implemented during the refurbishment outage, as one of the pre-conditions to reach the first criticality (January 2019).

In the case of CNA I the timeline was derived from an integral assessment of all measures identified for upgrading and/or making more robust the defence in depth concept. The integral assessment took due consideration of the safety significance of each proposed measure and was based on deterministic analysis, probabilistic safety assessment and engineering judgment.
As similar as CNE, the timeline had also connection with the LTO project. In this case three milestones can be identified:

1. For facing Phase A (already done),
2. During Phase A where a long outage is planned for next 2023.
3. For facing Phase B in next 2025.

It is important to stress that for both cases the timeline was developed in accordance with the safety significance of the upgrading measure and a prioritization of them was also done considering at which level of the defence in depth concept is aimed to.

No. 126
COUNTRY: CANADA
CNS-REF.-ART.: Article 7
PAGE OF REPORT: 41
CHAPTER OF NAT. REPORT: 3.7.4

"fines to be applied proportionately to the severity of the fault and as a function of the potential damage involved". How are the fines calculated? Is there a minimum/maximum monetary value?

The infractions are typified in different articles of the Sanctions Regime for Nuclear Power Plants; approved by Resolution Nº 63 (5/5/99) of ARN Board of Directors. Each type of infraction has associated a minimum and a maximum of monetary fine.

At the time of the issuance of the sanctions regime, different parameters such as the regulatory fee and the value of wages were taken into account to determine the values of fines. However, these values were fixed without updating so far.

On the other hand, a technical evaluation is performed at the time of imposing the amount of the fine. The technical evaluation involves analyzing the regulatory offences taking into account two parameters; 1) severity of the infraction and 2) potential of the damage. Likewise, these extremes are graduated in mild, moderate and severe. So if the infraction to the regulatory standard is considered to entail a severity of the serious infraction and the potentiality of the damage is also severe, then the amount of fine to be applied will be the maximum that is typified.

No. 127
COUNTRY: CANADA
CNS-REF.-ART.: Article 8
PAGE OF REPORT: 47
CHAPTER OF NAT. REPORT: 3.8.3.1.1

"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". Understanding that there is no control over the economic situation of the country, does the licensee or regulator have any plans in place to address this difficulty?

ARN had a severe loss of qualified personnel in the last years and has faced the situation of transferring personnel with competencies developed in sectors that carry out less priority activities, to sectors with higher priority activities. To address this issue in the near future, work is being done on the mapping of skills to develop an appropriate training plan that suits current staff and a knowledge management plan.

No. 128
COUNTRY: CANADA
CNS-REF.-ART.: Article 8
PAGE OF REPORT: 47
CHAPTER OF NAT. REPORT: 3.8.3.1.1

Good Performance: Providing technicians with in-depth radiation protection training (8 weeks, 7 hours a day) in addition to on-the-job training.

Argentina appreciates to Canada the identification of a good performance.

No. 129
COUNTRY: CANADA
CNS-REF.-ART.: Article 8
PAGE OF REPORT: 48
CHAPTER OF NAT. REPORT: 3.8.3.1.2

Radiological safety inspections and evaluations are conducted by Radiological Protection in Facilities and Practices division independent from licensing and control division. Does the Radiological Protection division share their results with LCNRD?

Yes, Radiological Protection division shares their results with LCNRD. As the division of responsibilities imposed by ARN´s structural organization states, Radiological Protection performs its activities in response of work packages asked for LCNRD. The results of their activities are internally submitted to LCNRD for their further management.

No. 130
COUNTRY: CANADA
CNS-REF.-ART.: Article 8
PAGE OF REPORT: 50
CHAPTER OF NAT. REPORT: 3.8.3.1.5

The UCE is working to formalize OJT. Is there an expected date for completion of this formalization?

As we stated in the reference document, the Education and Training Unit (UCE) is working on the formalization of the activities considered as OJT. As the UCE understands that this task is a constant updating process, this work is always "in progress". However, regarding to the establishment of procedures for the registration of these activities and their particular features, a pilot mechanism is currently running.

No. 131
COUNTRY: CANADA
CNS-REF.-ART.: Article 8
PAGE OF REPORT: 54
CHAPTER OF NAT. REPORT: 3.8.6

"The mentioned Law sets the amount of the annual regulatory fees, as a function of the nominal power installed for each NPP". Does this amount change with time to cover increased costs due to inflation over the years?

Art. 26 of the National Law of Nuclear Activity No. 24.804 states the following: "…In the case of nuclear power plants this annual regulatory fee may not exceed the value equivalent to the average annual price of one hundred megawatt hour (100 MW / h) in the Electricity Market determined based on the prices in said market corresponding to the previous year. Said fee must be paid per nominal installed megawatt during the lifecycle of the plant until the activities of removal the spent fuel from the reactor are completed…"

TRAP = PMAeE [$/MWH] x SPEB [MWH]

Where:

TRAP: Annual regulatory fee.

PMAeE: Estimated annual average price of the energy.

SPEB: Total gross power installed of the NPPs in operation.

That is, there are two variables that are involved in determining the value of the regulatory fee for year:

- The average annual price of 100 MWh in the Electricity Market.
- The nominal installed nuclear power.

The use of them results in the variation of costs over the years.

No. 132
COUNTRY: CANADA
CNS-REF.-ART.: Article 12
PAGE OF REPORT: 79
CHAPTER OF NAT. REPORT: 3.12.1.3.3

"Self-assessments program is carried out annually". Is there a schedule for which divisions/ programs are checked? If not, how is this decided upon?

A program of self-assessments with different objectives is designed annually, according to the need of each department.

No. 133
COUNTRY: CANADA
CNS-REF.-ART.: Article 14
PAGE OF REPORT: 98
CHAPTER OF NAT. REPORT: 3.14.3.1.3.1

The measurements for cooling the outer side of the pressure vessel and venting the containment will be evaluated in due course. Are there dates planned for the initiation of these measurements?

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 were analysed and extra efforts had to be made to adapt codes to the Atucha reactors. 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). In the past years, preliminary results were obtained with RELAP5 / SCDAP. These calculations were followed by more complex analysis with ANSYS / CFD code, performed for CNA II NPP. The results of these analyses showed that the countermeasure is not successful in a scenario of LOCA in the moderator circuit with failure of safety injection system or in a SBO scenario. Based on these results, it was decided to rule out this countermeasure for Unit II and Unit I. It should be noted that the results for Unit II are extrapolable to Unit I in this case. As it was mentioned a parallel course of analysis has been started, to assess possible stabilization of molten material inside sump, to avoid an early containment breach, so as to decrease consequences in public as far as reasonable achievable. This task is being performed as part of phase B of CNA I LTO. This project is part of the Conceptual Improvement Plan that will be presented for consideration to ARN in March 2020.
Regarding the venting filtered containment system it is also planned to be implemented as part of phase B of CNA I LTO. (see section 3.6.5.2).

No. 134
COUNTRY: CANADA
CNS-REF.-ART.: Article 17
PAGE OF REPORT: 142
CHAPTER OF NAT. REPORT: 3.17.2.3.2.2

A seismic re-evaluation is planned as part of the CNE life extension project. Has this re-evaluation been conducted now that CNE has re-started?

Seismic re-evaluation was carried out in the initial stages of the life extension project and allowed the realization of several design improvements as described in the National Nuclear Safety Report 2019, 3.17.2.3.2.2.1. Earthquakes.

No. 135
COUNTRY: CANADA
CNS-REF.-ART.: Article 17
PAGE OF REPORT: 144
CHAPTER OF NAT. REPORT: 3.17.2.3.2.2.1

A 24 inch rupture disc assembly shall be installed on the top of the existing calandria vault inspection port. It is foreseen to be implemented by the end of 2017. Was this addition completed as scheduled?

This design modification was implemented during the life extension project as scheduled and assured the pressure relief capability of the calandria vault during a severe accident. This rupture disk breaks when the internal pressure reaches 69 kPa and was sized to evacuate the mass of steam arising from a severe accident.

No. 136
COUNTRY: CANADA
CNS-REF.-ART.: General
PAGE OF REPORT: 2-3
CHAPTER OF NAT. REPORT: 1.3

The fourth NPP will be of a different design than the first 3 NPPs. Why was this SMR design chosen? When will the new reactor be commissioned?

As it is stated in the Section 1.3 of the National Report, the National Government promotes nuclear activities in the country.

Within this framework, the National Congress, through Law No. 26,566 declared of national interest the design, implementation and commissioning of the CAREM prototype reactor being built in Argentina, committing CNEA for that purpose.

No. 137
COUNTRY: CANADA
CNS-REF.-ART.: General
PAGE OF REPORT: 8
CHAPTER OF NAT. REPORT: 1.4.2.1

How are these events (large LOCAs, LOFA, control rod ejection) eliminated via this CAREM 25 prototype design? Please explain how risk reduction systems operate to accomplish this.

The CAREM design reduces the number of sensitive components and potentially risky interactions with the environment.

CAREM is a natural circulation based indirect-cycle reactor with features that simplify the design and improve safety performance. Its primary circuit is fully contained in the reactor vessel and it does not need any primary recirculation pumps. The self-pressurization is achieved by balancing steam production and condensation in the vessel, without a separate pressurizer vessel.

Regarding to Large LOCA events, the limitation of the diameter of the connection to the Reactor Pressure Vessel (RPV) to 38.1 mm and the penetrations of the pressure vessel at the highest possible level, above the active area of the GV, which reduces the impact of events due to loss of refrigerant in the reactor and in the requirements of safety systems.

Regarding the Control Rod Ejection event, is a typical initiating event of pressurized water-cooled reactors that were excluded in the safety analysis because the Control Rod mechanisms are hydraulic and in-vessel (internal of RPV) without being subjected to pressure differences.

Risk reduction systems perform functions at level 2 of DiD. These systems will be considered important for safety and will be assigned a degree of safety classification to provide a level of protection before level 3A, trying to avoid the thresholds of activation of the first protection system in relation to the demand for Safety Systems .

No. 138
COUNTRY: CANADA
CNS-REF.-ART.: Article 6
PAGE OF REPORT: 24
CHAPTER OF NAT. REPORT: 3.6.5.1

Aging Assessment - Argentina should consider a benchmarking of the Aging Management plans for example, those developed in other CANDU countries.

Argentina appreciates the suggestion and will consider performing a benchmarking of the Ageing Management plans. However it is worthwhile to mention that Argentina is an active member of IAEA-IGALL group which provides a good platform for sharing experiences, included CANDU experience in that regard.

No. 139
COUNTRY: CANADA
CNS-REF.-ART.: Article 6
PAGE OF REPORT: 27
CHAPTER OF NAT. REPORT: 3.6.5.1

Has the management of SG component replacement considered operating experience from other CANDU installations?

All design changes implemented for the new Steam Generators (SG) were due to 3 fundamental factors:

- Plant repowering
- Seismic requalification
- Operational experience

Due to repowering, the thermal transfer area was increased through the implementation of longer tubes.

To comply with the new seismic requirement, design of the Quinshan steam generators was adopted as reference, which implied changes in the internal support of the pipe harness and certain modifications in the anchors of the external support on the SG shell.

In relation to the design changes associated with the operating experience, the new design and type of material for the tube support plates and the U-bend supports can be mentioned. The new support design is of the "Flat bar" type and the adopted material is 410 S stainless steel.

In addition, an inspection port in the U-bend region, 3 support plate inspection ports, 2 new inspection ports in the pre-heater region and 1 additional waterlancing port were added.

Carbon steel alloy subcomponents that are part of the pressure envelope, with wet surfaces in service, have 0.2% minimum Cr in their chemical composition.

No. 140
COUNTRY: CANADA
CNS-REF.-ART.: Article 8
PAGE OF REPORT: 49
CHAPTER OF NAT. REPORT: 3.8.3.1.5

What is the curriculum based on for the ARN postgraduate courses that the ARN dictates in academic partnership with the School of Engineering of the University of Buenos Aires (FIUBA)?

The curriculum of the Specialization Degree in Radiation Protection and Security of Radiation Sources and the Specialization Degree in Nuclear Safety, are based on the elements developed by the standard syllabus developed by the IAEA for PGEC and on the document developed by the IAEA for BPTC, but also adds some particularities and deepens in some specific topics.

No. 141
COUNTRY: CANADA
CNS-REF.-ART.: Article 16
PAGE OF REPORT: 126
CHAPTER OF NAT. REPORT: 3.16.6

"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" - Could you please elaborate on the status of this work?

During 2019, ARN and the Secretariat of Civil Protection of the Nation signed a notification agreement for the SINAGIR Public Alert Platform, whereby emergency situations are disseminated and notified to the public. In addition, ARN has access, through the Secretariat, to integrated computer communication and information exchange systems where the characteristics of the emergency and the available resources of all the response organizations involved are communicated. On the other hand, the efforts to develop an updated national plan for nuclear emergencies began.

No. 142
COUNTRY: CANADA
CNS-REF.-ART.: Article 18
PAGE OF REPORT: 159
CHAPTER OF NAT. REPORT: 3.18.3.5

"Despite 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 the requisites set by NA-SA and applicable laws, regulations, standards, specifications and any other requirements established by ARN" - Could you please elaborate on the status of this? What managed process will be used and how does it differ from those used on the first 3 NPPs?

According to the agreed draft version of the Contract for the Fourth NPP, the Contractor shall be responsible for the design of the plant, which shall be reviewed and accepted by NA-SA. The Management System shall be established by the Contractor and approved by NA-SA and it shall fulfill the requirements set out by ARN’s regulations, IAEA Guidelines, NA-SA´s requirements as well as ISO and other applicable standards.

The Management System for the Fourth NPP Project is a logical evolution from the Quality Management System used for Argentina´s previous three NPP Projects that have been carried out during the 70´s and 80´s. In particular it takes advantage on the lessons learned during the application of the one recently used for the completion of the Atucha II NPP. It has been improved and updated according to the last versions of the applicable documents, codes and standards.

No. 143
COUNTRY: RUSSIAN FEDERATION
CNS-REF.-ART.: General
PAGE OF REPORT:
CHAPTER OF NAT. REPORT: General

What is Argentina's response to the "Major Common Issues Arising from Country Groups Discussions" (paragraphs 25 to 34 of the Summary Report of the 7th Review Meeting of the Contracting Parties to The Convention on Nuclear Safety)?

The Argentina´s response to the “Major Common Issues Arising from Country Group Discussions” is embedded into the different sections of the National Report.

As example, it can be mentioned that concerning safety culture, section 3.10.2.2 “SAFETY CULTURE AND ITS DEVELOPMENT” provides a description of the activities performed in that regard. Also, it is important to mention that Argentina is actively participating in a Regional Latin America (RLA) project from the IAEA with the purpose of supporting NPP life management and safety culture practices. This project contemplates the participation of the regulatory body as well as the operator and its biennial plan is developed according to the needs of each participant organization.

Under this project several activities were performed, as for example:

- Preparation for peer review mission like SALTO (Safety Aspects of Long Term Operation).
- Preparation of OSART (Operational Safety Review Team) mission.
- Supporting assessments and development of continuous improvement programs on operational safety, leadership and culture for safety.
- Practical support to ensure strong leadership and culture for safety during changes.
- To strengthen the implementation of nuclear oversight in the regulatory body.

Regarding the issue “International Peer Reviews”, the National Report describes the activities performed in the frame of Ibero-American Forum of Radiological and Nuclear Regulatory Agencies (FORO) which illustrates the Argentina’s commitment on the participation in international peer reviews and exchange of information.

Communication to stakeholders by ARN was improved relaunching an updated ARN website at www.argentina.gob.ar/arn with an enhanced a content, tailored to 3 different users’ profile –general public, regulated and students, in a more modern and accessible website for people with different abilities, and with responsive design.

In summary, it can be concluded that “Major Common Issues Arising from Country Group Discussions” were considered, to the applicable extent, as an opportunity for improving activities of the regulatory body or operators.

No. 144
COUNTRY: RUSSIAN FEDERATION
CNS-REF.-ART.: Article 11
PAGE OF REPORT:
CHAPTER OF NAT. REPORT: Section 3.11

What actions is Argentina taking to ensure the availability of financial resources in the case of a radiological emergency?

ARN has a permanent financial fund to meet the first needs in case of a radiological emergency.

No. 145
COUNTRY: RUSSIAN FEDERATION
CNS-REF.-ART.: Article 16
PAGE OF REPORT:
CHAPTER OF NAT. REPORT: 3.16

Are there unannounced emergency drills and exercises in Argentina? If yes, then what are the lessons learned from such exercises compared to planned exercises and exercises?

In the case of NPPs, exercises involving the population and response organizations are planned and announced activities.

No. 146
COUNTRY: ITALY
CNS-REF.-ART.:
PAGE OF REPORT: Annex II - page 2
CHAPTER OF NAT. REPORT: Annex II

With reference to the meeting held in Argentina in 2017 on supply chain and contractors qualification, could Argentina provide information about the current national situation? Have operator's difficulties for the supply of components having relevance for safety? Is the aging management affected from any in the supply capabilities?

The company owner the NPPs has not had relevant difficulties to meet the requirements of the plants in terms of important safety supplies.

On the other hand, it has not been observed that aging management was affected by lack of supplies.

No. 147
COUNTRY: ITALY
CNS-REF.-ART.: Article 6
PAGE OF REPORT:
CHAPTER OF NAT. REPORT: 3.6.3.1.3

Could Argentina explain the countermeasures put in place to avoid components CFSI (Counterfeit, Fraudulent and Suspect Items)? Were there in the past any finding? If yes, what were the consequences on NPPs operation?

On the basis of a graded approach to safety, a reception committee in each plant is responsible for verifying that the technical specifications of the received items comply with those stated in the purchase order. Also, the items are stocked in controlled conditions according to the specifications. Before installation in the plant, items which do not comply with defined standards are clearly identified and segregated to prevent inadvertent use.

No. 148
COUNTRY: ITALY
CNS-REF.-ART.: Article 8
PAGE OF REPORT: 45
CHAPTER OF NAT. REPORT: 3.8.3

With reference to human resources at the ARN, could Argentina provide information on the staff age and, if needed, on the expected recruitment to assure staff turnover?

The staff age is distributed in the following segments:

Ages over 60 years: 15%

Ages between 50 and 60 years: 18%

Ages between 40 and 50 years: 22%

Ages between 30 and 40 years: 39%

Ages under 30 years: 6%

ARN had in the last years a severe loss of qualified personnel because of the retirement age. To address this difficulty in the future, work is being done on the mapping of skills to develop an appropriate training plan that suits current staff and a knowledge management plan.

No. 149
COUNTRY: ITALY
CNS-REF.-ART.: Article 11
PAGE OF REPORT: 68
CHAPTER OF NAT. REPORT: 3.11

Could Argentina provide information about financial resources with regard to remediation following possible nuclear accidents and to ensure decommissioning activities?

Nuclear accidents: Argentina contemplates this issue in Law 25313 (Protocol to Amend the Vienna Convention on Civil Liability for Nuclear Damage and Convention on Supplementary Compensation for Nuclear Damage).

Article 9 - Every natural or legal person to develop a nuclear activity must:

(…)

c) Assume the civil responsibility that for the operator of a nuclear installation determines the Vienna Convention on Civil Liability for Nuclear Damage, ratified by law 17,048, for the sum of eighty million US dollars (US $ 80,000,000) per accident nuclear in each nuclear installation. It must be covered by insurance or financial guarantee to the satisfaction of the National Executive Power or its designee, assuming the National State the remaining responsibility.

Decommissioning: In accordance with the provisions of Article 2, subsection e) of Law No. 24,804, it is the responsibility of the National Atomic Energy Commission to determine the manner of decommissioning of nuclear power plants; the scheme presented has not yet been approved.

In accordance with the provisions of PEN Decree No. 1,390/98 regulating Law No. 24,804 on nuclear activity, the fund with the necessary resources to face the decommissioning from each nuclear power plant would be created with the contributions of the company that became an operator of nuclear power plants to be privatized. Law No. 26,784, in its article 61, repeals article 34 of Law No. 24,804, so NA-SA is no longer subject to privatization. For this reason and in accordance with the provisions of article 37 of Law No. 24,065, which states that "the generation and transportation companies of total or majority ownership of the national State will have the right to recover only their total operating and maintenance costs that allow to maintain the quality, continuity and safety of the service…" the responsibility of financing the decommissioning of nuclear power plants is not assumed to date by NA-SA.

Similarly, it is noted that to date the sixth National Report 2017 of the "Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management" has been issued. In its point F.6.5 it states that, the financing for the constitution of said fund must be assumed by the national State with own funds.

No. 150
COUNTRY: ITALY
CNS-REF.-ART.: Article 11
PAGE OF REPORT: 71
CHAPTER OF NAT. REPORT: 3.11.2.3

With reference to human resources at the NA-SA, could Argentina provide more information on the sufficiency of the number of qualified staff and on recruitment plans, if needed?

Yes, more information can be given if needed. We understand that we have the qualified staff as expressed in other answers. As also stated in other answers, recruitment plans are included within the Staffing Plan that is discussed and agreed upon with the Plant Managers.

No. 151
COUNTRY: ITALY
CNS-REF.-ART.: Article 15
PAGE OF REPORT: 113
CHAPTER OF NAT. REPORT: 3.15.1

Could Argentina clarify how and by which organization all the information concerning the levels of radioactivity in the environment around the plant are made available to the public?

The information concerning the levels of radioactivity in the environment is made available to the public by ARN.

Each year, ARN publishes the Annual Activity Report that compiles the main tasks carried out in the areas of radiological and nuclear safety, safeguards and physical protection including the levels of radioactivity in the environment.

The Annual Activity Report is prepared in accordance with the provisions of article 16 of the National Law of Nuclear Activity - Law No. 24804 and fulfills the function of informing stakeholders.

No. 152
COUNTRY: ITALY
CNS-REF.-ART.: Article 15
PAGE OF REPORT: 113
CHAPTER OF NAT. REPORT: 3.15.1

Could Argentina explain how and by which organization independent environmental monitoring is performed around the plant?

ARN carries out an Environmental Plan for Radiological Monitoring (in Spanish, PMRA) in the surroundings of the nuclear power plants involving sampling and subsequent measurement of radionuclide concentration in various environmental matrices: water, air, soil, sediments and food.

This monitoring is independent of those carried out by the Primary Responsible of the plant.

Samples are processed and evaluated in ARN's laboratories, located in the Ezeiza Atomic Center (CAE). These laboratories are dedicated to sample pretreatment, gamma spectrometry, tritium measurement, determination of uranium by fluorimetry and kinetic phosphorescence, determination of strontium, measurement of alpha emitters and beta, radon measurements, among others. The techniques used in these laboratories are accredited under the standard IRAM 301:2005 (ISO / IEC 17025: 2005).

No. 153
COUNTRY: ITALY
CNS-REF.-ART.: Article 15
PAGE OF REPORT: 122
CHAPTER OF NAT. REPORT: 3.15.7

Could Argentina provide information on what are, according to the radiation protection laws, the dose limits for members of the public during normal and accident conditions? Do, in normal conditions, the calculated doses take into account the exposure coming from direct external irradiation and doses from gaseous and liquid effluents?

Standard AR 10.1.1 establishes.

The dose limits for the public are:

a. an effective dose of 1 mSv per year; and in special circumstances, a highest effective dose value in a single year, provided the average dose effective for five consecutive years does not exceed 1 mSv
b. an equivalent dose in lens of the eyes of 15 mSv per year
c. an equivalent dose in skin, hands or feet of 50 mSv per year.

For these cases, Standard AR 3.1.3 “Criterion Curve”, establishes a limit to the risk from any accident situation through a probabilistic quantification.

Moreover, Standard AR 10.1.1 establishes:

In the design of an installation Class I, the Responsible Entity shall ensure that the annual probability of occurrence of a postulated accidental sequence is verified (or category of release), estimated according to requirement 47, with potential radiological implications on the public, and the effective dose estimate resulting from that sequence on the representative person do not define a point in the unacceptable area of Figure 2.

Figure 2 is the “Criterion Curve for the public”.

In addition, Standard 10.1.1 establishes:

EMERGENCY EXPOSITION SITUATION

Public exposure

The protection strategies should be such that for the residual dose reference levels, expressed in effective dose, are applied in the range of 20 to 100 mSv.

In normal situations, the doses calculated for public include those due to gaseous and liquid discharges as well as direct doses by external irradiation, if applicable.

No. 154
COUNTRY: ITALY
CNS-REF.-ART.: Article 15
PAGE OF REPORT: 117
CHAPTER OF NAT. REPORT: 3.15.5

Could Argentina provide information if Radiation Protection Laws establish specific provision for external workers liable to be exposed to ionizing radiation?

ARN implements radiation protection through regulatory standards.

The Radiological Protection criteria adopted by the Regulatory Body to control the dose received by workers is consistent with ICRP’s recommendations. Regulatory Standards AR 3.1.1, AR 4.1.1 and AR 6.1.1 applied to nuclear power reactors, research reactors and radioactive facilities Type I set different criteria to ensure that the occupational dose to workers stays as low as reasonably achievable and lower than the established dose constraints and there is not a distinction about permanent or external workers.

Occupational Dose Limits

Dose limits for workers are as follows: The effective annual dose limit is 20 mSv. This value shall be considered as the average in 5 consecutive years (100 mSv in 5 years), not exceeding 50 mSv in any single year.

Regulatory standard AR 10.1.1 sets that employees are responsible for their compliance with the procedures established to ensure their own protection as well as the protection of other employees and of the public. This condition is consistent with the recommendations of the International Atomic Energy Agency (IAEA).

No. 155
COUNTRY: ITALY
CNS-REF.-ART.: Article 16
PAGE OF REPORT: 125
CHAPTER OF NAT. REPORT: 3.16.4

Could Argentina clarify which organisation(s) is (are) responsible for providing information to the public both prior and in case of a nuclear emergency?

Through the recently signed agreement between ARN and the Secretary of Civil Protection of the Nation, there is an alert platform in which basic information is disseminated in case of emergencies about risks and recommended measures, so that the public is also protected. In addition, the population remains informed during it.

On the other hand, the populations surrounding nuclear power plants have a public early warning system.

No. 156
COUNTRY: ITALY
CNS-REF.-ART.: Article 16
PAGE OF REPORT: 128
CHAPTER OF NAT. REPORT: 3.16.8

How are information to neighboring Countries provided in emergency preparedness and emergency situation?

Argentina has ratified Early Notification and Assistance Conventions in case of Nuclear Emergencies through National Law 23731. Notification to the IAEA and the member countries is through the USIE platform.

No. 157
COUNTRY: ITALY
CNS-REF.-ART.: Article 18
PAGE OF REPORT: 150
CHAPTER OF NAT. REPORT: 3.18.3.1.1

During past 7th review process, Argentina specified that containment venting filtration system was under assessment at CNA I & II NPPs. Could Argentina clarify the reasons for which the assessment of the filtered containment venting system has not been completed yet, and when is its implementation foreseen?

Independent filtered containment venting systems for CNA I & II are planned to be implemented at the same time in order to take the advantages of carrying out similar projects together. Its implementations are foreseen for phase B of CNA I LTO. (see section 3.6.5.2)

No. 158
COUNTRY: ITALY
CNS-REF.-ART.: Article 16
PAGE OF REPORT: 129
CHAPTER OF NAT. REPORT: 3.16.9

Among different computational tools that operate at the Emergency Control Center of ARN is there a system able to acquire plant parameters in order to evaluate the source term?

Currently, ARN in its facilities does not have a computer system with required safety conditions to access the NPP plant information.

No. 159
COUNTRY: ITALY
CNS-REF.-ART.: Article 14
PAGE OF REPORT: 93
CHAPTER OF NAT. REPORT: 3.14.3

For existing installations could Argentina provide a list of adopted safety improvements to meet the VDNs objective of avoiding early radioactive releases or radioactive releases large enough to require long term protective measures?

As it is stated in Article 6 of the National Report, 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.

Detail of the activities performed during the reported period can be found in section 3.6.3, “Actions leading to safety improvements” and in section 3.6.4, “Improvement activities”, of the National Report.

No. 160
COUNTRY: ITALY
CNS-REF.-ART.: Article 17
PAGE OF REPORT: 134
CHAPTER OF NAT. REPORT: 3.17.2.3

How frequently and according to which procedure site related factors are re-evaluated with particular reference to natural hazards?

In Argentina it is required for all NPP the development of a Periodic Safety Review (PSR) every ten calendar years, according to the IAEA SSG-25, “Periodic Safety Review for Nuclear Power Plants”. As part of this PSR, Safety Factor 1: “Plant Design” and Safety Factor 7: “Hazard Analysis”, requires a site re-assessment in light with the latest regulations/practices and modern standards. In addition, Argentina is reactive from events like Fukushima.

No. 161
COUNTRY: ROMANIA
CNS-REF.-ART.: Article 14
PAGE OF REPORT: 13
CHAPTER OF NAT. REPORT: 2.6

Please provide more information on the new events for which it is mentioned that the trip coverage has been improved.

The new events for which the trip coverage has been improved are:

- Loss of Moderator Heat Sink
- Loss of Moderator Circulation

For those events, there were no effective trip parameters installed on shutdown systems at Embalse. Coverage was provided by regulating system actions only.

After refurbishment, the effective trip parameter for both shutdown systems is “High Moderator Level”. Anyway, the fuel channel integrity is not compromised for these events. The only potential dose comes from tritium released from the moderator inventory spilled out of the Calandria. The quantity of tritium released is expected to be small. The release would be well within regulatory limits. This is not a limiting acceptance criterion for shutdown system effectiveness, which is primarily determined by prevention of fuel failures.

The main trip coverage´s improvements of the reactor´s shutdown systems can be summarized as follow:

1) Events that were not covered by at least one trip parameter of each shutdown system and are now covered by at least one trip parameter of each shutdown system:
    - Single main HTS pump trip
    - Single main HTS pump seizure
    - Feedwater line breaks
    - Steam line breaks outside containment
    - Loss of feedwater flow to one boiler
    - Loss of moderator inventory
2) Events that were not covered by at least one trip parameter of each shutdown system and are now covered by two or more trip parameter of each shutdown system:
    - In-Core LOCA
    - Loss of Class IV power (external power)
3) Events that were covered by only one trip parameter of each shutdown system and are now covered by two or more trip parameter of each shutdown system:
    - Large LOCA
    - Small LOCA
    - Loss of primary circuit pressure an inventory control: depressurization
    - Loss of service water

No. 162
COUNTRY: ROMANIA
CNS-REF.-ART.: Article 18
PAGE OF REPORT: 14
CHAPTER OF NAT. REPORT: 2.6

Please provide more information on the modifications implemented to improve the reliability ECC, referring to the extra lines for injection, relocation of sump level sensors to improve measurement and the reduced frequency for the interface LOCA event through the ECCS line. What is the basis for these modifications (new safety analyses, operational experience, etc.)?

The design changes to the ECC were mainly to meet the new objectives of the PSA in terms of probability of severe damage to the core. Thus, it was proposed to increase the reliability and availability of the system.

The main ECC modifications were the following:

- Increase the size of valves and associated pipes corresponding to the recirculation of the emergency cooling system (PV23 / PV24). This modification was made to provide an alternative path for medium pressure injection from the Dousing tank to the aspiration of the ECC pumps.
- Replacement of the manual valve V7 for manual check valve. This eliminates the need for manual operator action in case of Design Bases Earthquake. In this case, the long-term replenishment of water to PHTS (Primary Heat Transport System) from the Dousing tank or from the Emergency Water Supply is done through MV75 and the V7 valve is required to be closed.
- Installation of redundant valves to the Dousing tank isolation valves to ensure the isolation of the Dousing tank at the end of the medium pressure stage avoiding the entry of air into the suction of the ECC pumps.
- Automatic transfer of medium pressure stage to the low pressure stage, avoiding the need for operator intervention.
- New level measurements of the Dousing tank were provided to ensure accurate measurement.
- Seismically qualified electric power supply for ECC pumps. This design change was to ensure the operation of the ECC pumps in the case of LOCA and 24 hours after a site earthquake (SDE) (by not considering Class III Diesel available).
- Provision of cooling water to the ECC heat exchanger from the Emergency Water Supply (EWS) in case of loss of Service Water System.
- Additional alarms to keep the operator informed. For example, low pressure alarm in the ECC water tanks and the relocation of the sump level sensors because they were close to the suction of the ECC pumps and could cause wrong measurement.

No. 163
COUNTRY: SOUTH AFRICA
CNS-REF.-ART.: Article 17
PAGE OF REPORT:
CHAPTER OF NAT. REPORT: 3.17

As part of evaluating the radiological impact of NPP operation on the public and environment, does Argentina monitor the prevalence of cancer among population groups living around NPP sites (for example, by conducting relevant research studies)?

Argentina as part of the requirements does not have Ecology, and fresh water supply. Is there any particular reason as to why?

1) There are no specific studies on the prevalence of cancer in the population around the NPPs. To evaluate the environmental radiological impact, monitoring and follow-up of authorized discharges to the environment are carried out, sampling and measurements of radionuclides are carried out in the different environmental matrices. Likewise, the dose to the representative person of each site is calculated for the emitted discharges.
2) Standard AR 10.10.1 “Site evaluation for NPP” establishes specific requirements and criteria for this purpose:

The environmental radiological impact must be evaluated considering all the operating states and accident conditions, including those cases that may lead to emergency measures.

The surrounding geographical area should be evaluated considering the foreseeable present and future characteristics and the distribution of the population, including the present and future uses of land and water, and any other characteristic that may affect the possible consequences of radioactive emissions to the public and the environment.

Land and water uses of the site area

Land and water uses should be characterized to assess the effects of the nuclear power reactor on the site area and to prepare emergency plans. The evaluation should include land and water bodies that can be used by the public or can serve as a habitat for organisms present in the food chain.

Possible effects on the public should be evaluated due to the dispersion of radioactive materials, both in surface waters and groundwater, using the data and information collected.

Environmental radioactivity

Before commissioning a nuclear power reactor, the environmental radioactivity of the atmosphere, hydrosphere, lithosphere and biota in the site area must be determined, in order to assess the effects of the operation of the nuclear power reactor. The data obtained will constitute the environmental radiological baseline and should be collected periodically for a period of at least one year, before commissioning.

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 tiIter has the following functions in the indicated sequence:

- Take-over of the fuel bundle column with the spent fuel assembly.
- Removal of the decay heat by cooling with heavy water.
- Drying and cooling the spent fuel assembly with gas.
- Flooding and cooling of the tiIter with heavy water.
- Tilting to the horizontal position and connecting with the fuel transfer tube.
- Transfer of the fuel assembly into the fuel transfer tube.

When a new fuel bundle column is transported from the fuel pool building into the tilter via the transfer tube, and later from there into the refuelling machine, the process of cooling and change of cooling medium takes place in the reverse order.

The 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 ℃
Coolant channel outlet temperature	303.3 ℃
Maximum temperature on the fuel assembly cladding surface	325 ℃
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 ℃
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 ℃
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

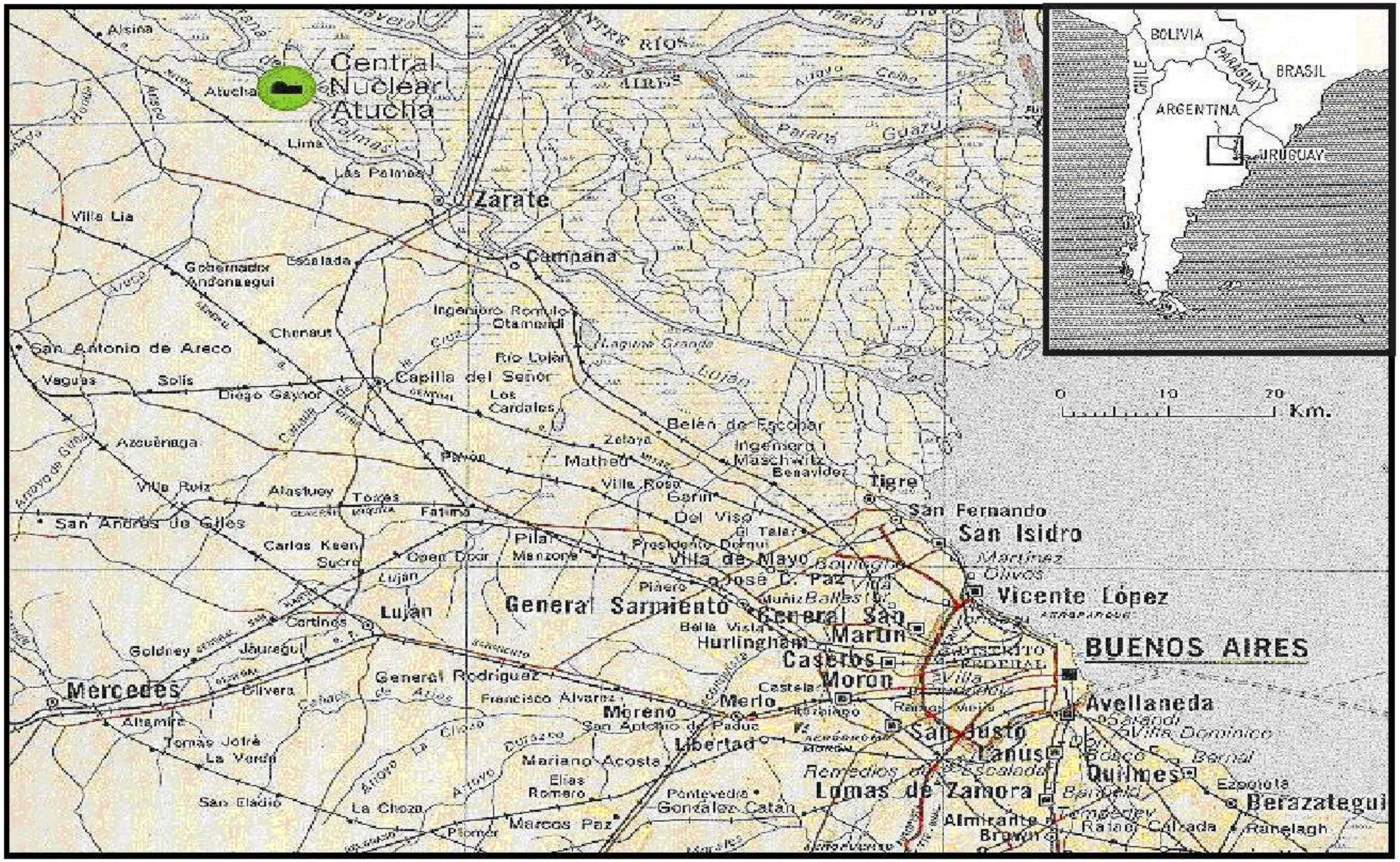


Figure III.1-1 - Atucha I Nuclear Power Plant - Geographic Location

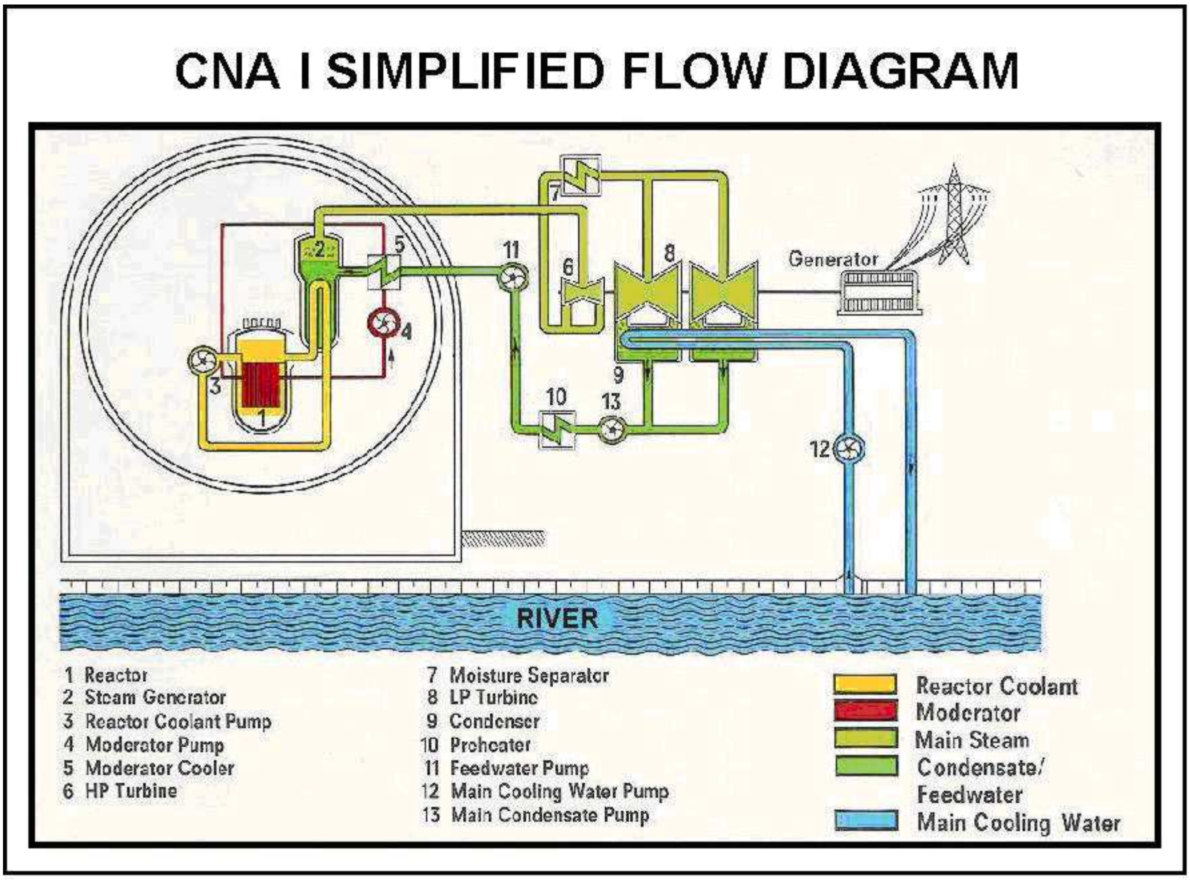


Figure III.1-2 - Atucha I Nuclear Power Plant - Main Systems

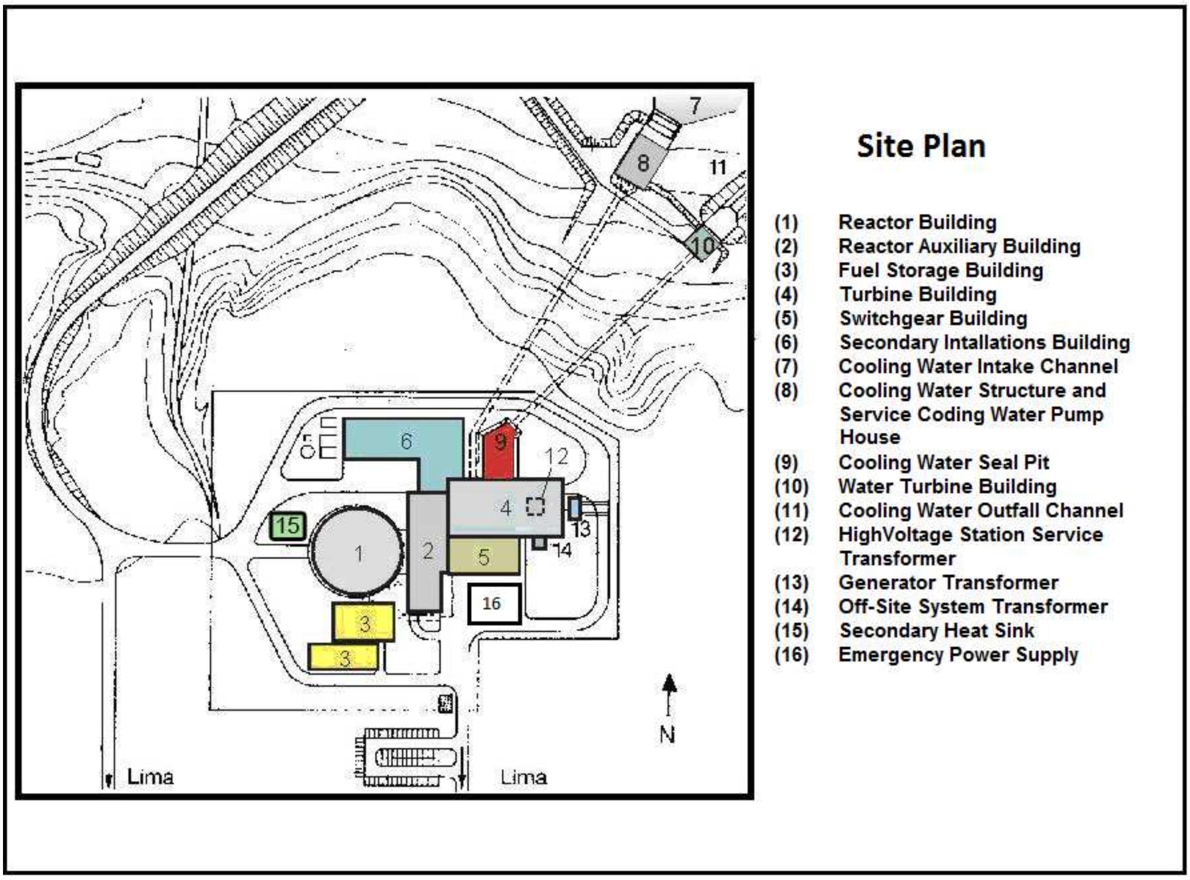
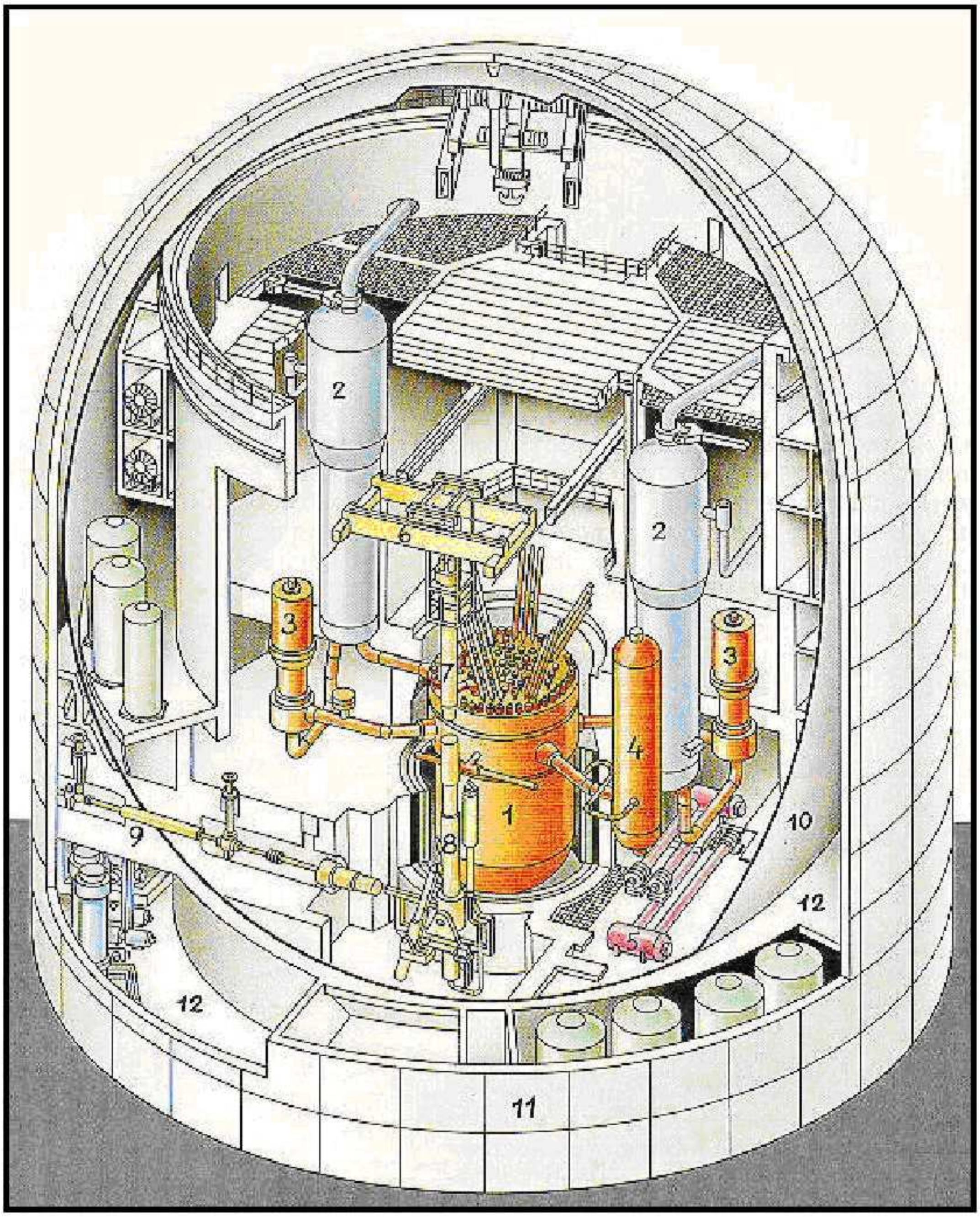


Figure III.1-3 - Atucha I Nuclear Power Plant - Main Building and Structures

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

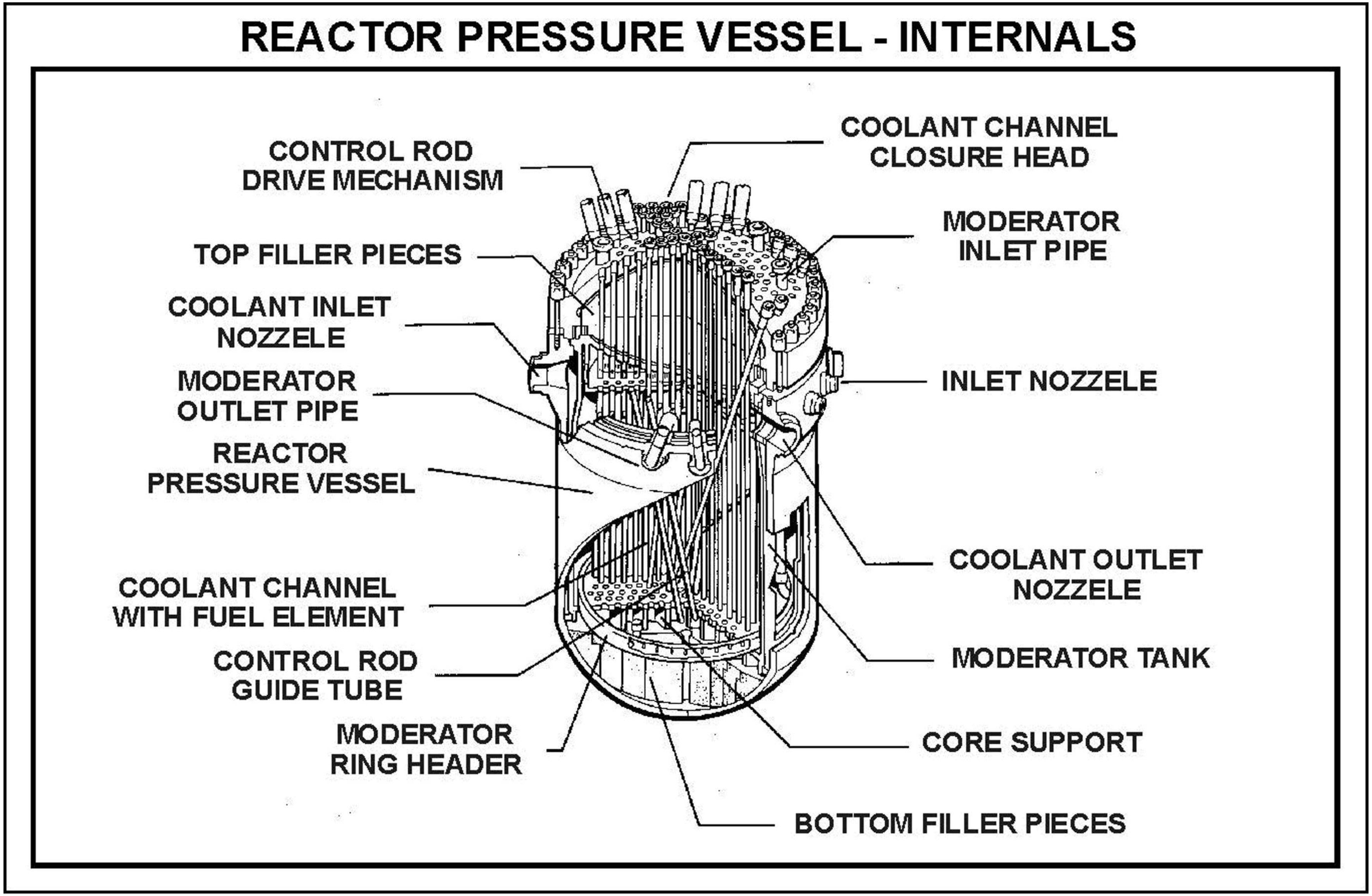
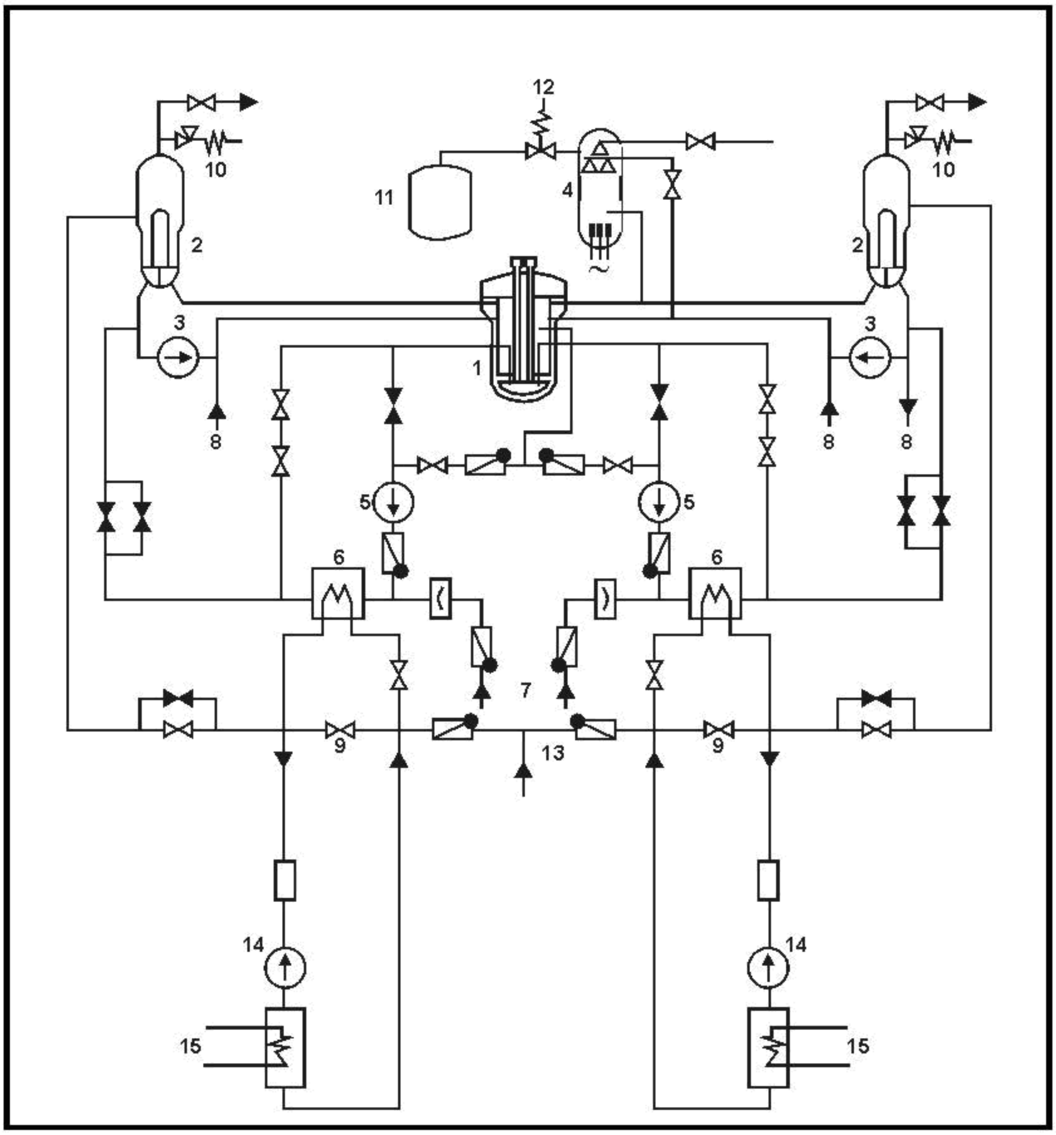


Figure III.1-5 - Atucha I Nuclear Power Plant - Reactor Pressure Vessel and Internals

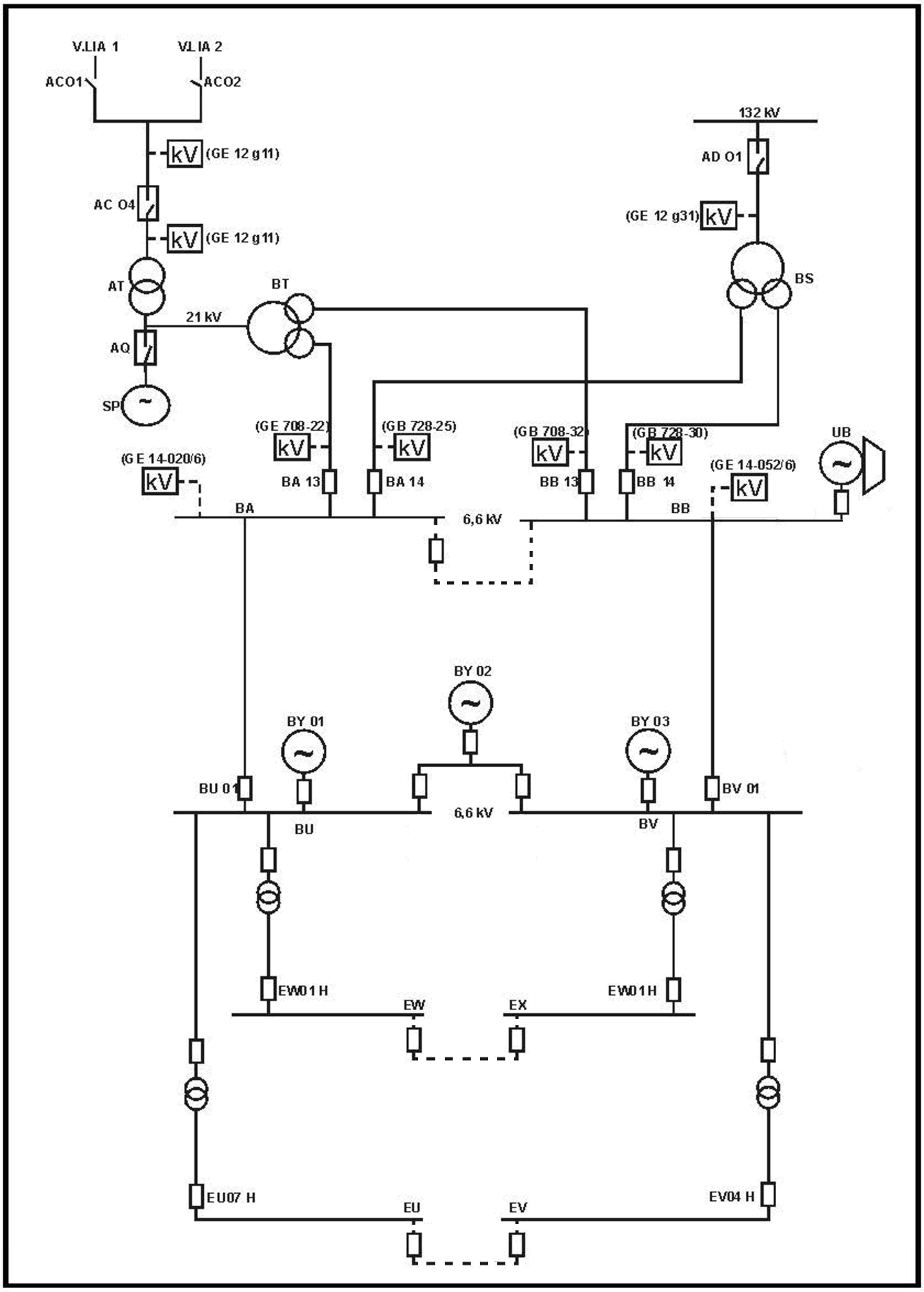
REACTOR COOLANT SYSTEM AND MODERATOR SYSTEM



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

BASIC ELECTRICAL POWER SYSTEM (SINGLE LINE DIAGRAM)



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

Figure 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 National Atomic Energy Commission of Argentina (CNEA) 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. *CNE completed its first operating cycle at the end of 2015. As identified and described in previous National Nuclear Safety Reports, many improvements were implemented during refurbishement outage carried out from 2016 to 2019, in order to extend the life of the plant for another 30 years of operation.*

CNE returned to service in January 2019, starting its second operating cycle.

At present, the owner of CNE is Nucleoeléctrica Argentina S.A., and the plant provides a net electric power of 608 MWe to the interconnected national system. *The works carried out within the framework of the Life Extension Project allowed CNE to increase its installed capacity by 8 MW.*

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 *2064* 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 building has reinforced concrete slabs and consists of one single building formed of concrete walls and roof. This building is divided into six rooms, four of which contain generator sets together with their auxiliary sistems and the other two house elecrical and battery rooms of the Diesel Generators. Partition walls between these rooms are full height reinforced concrete.

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 *380* 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	*608* MWe
Gross nominal electric power	*656* 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 ℃
Temperature in the reactor outlet header	310 ℃
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

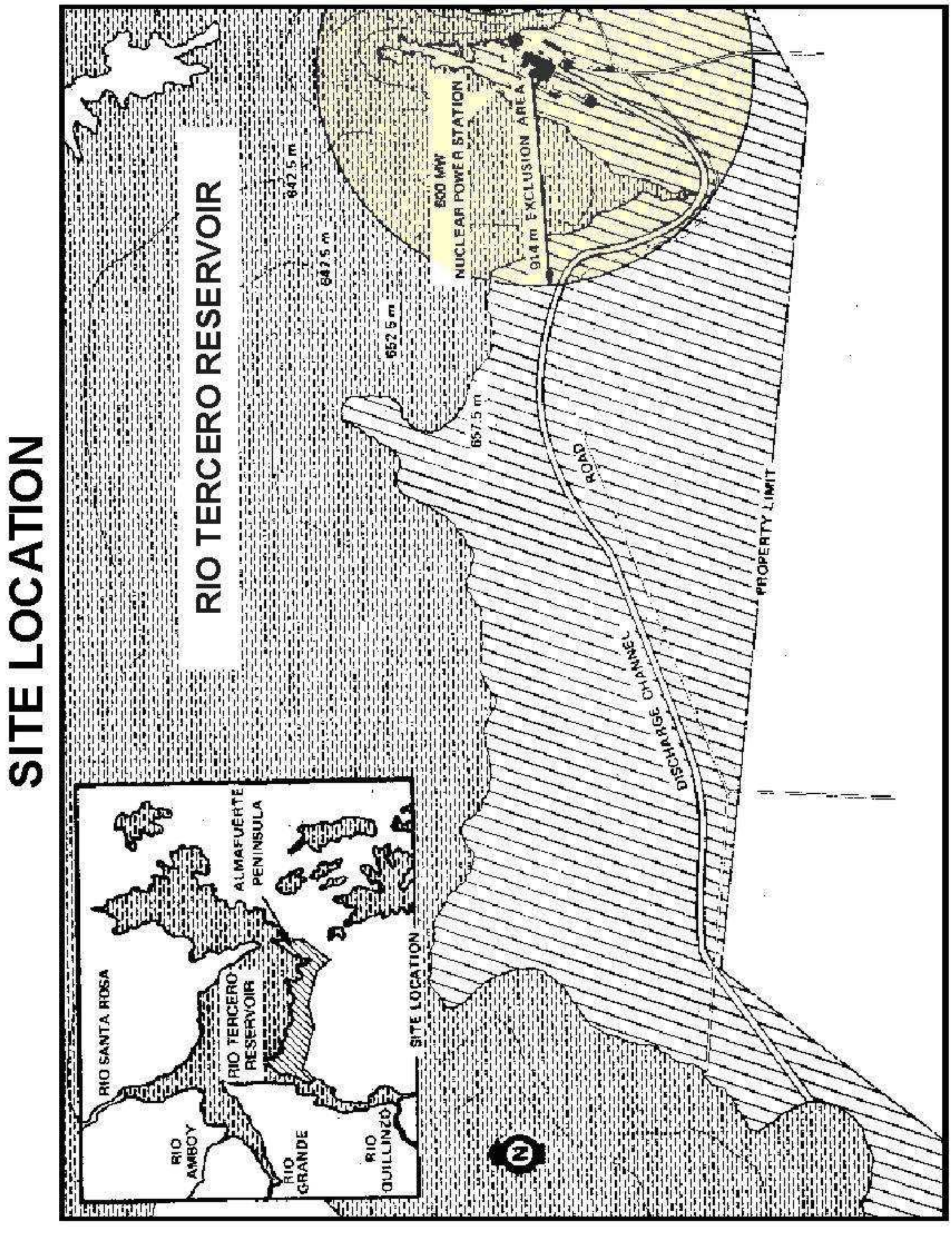


Figure III.2-1 - Embalse Nuclear Power Plant - Site Location

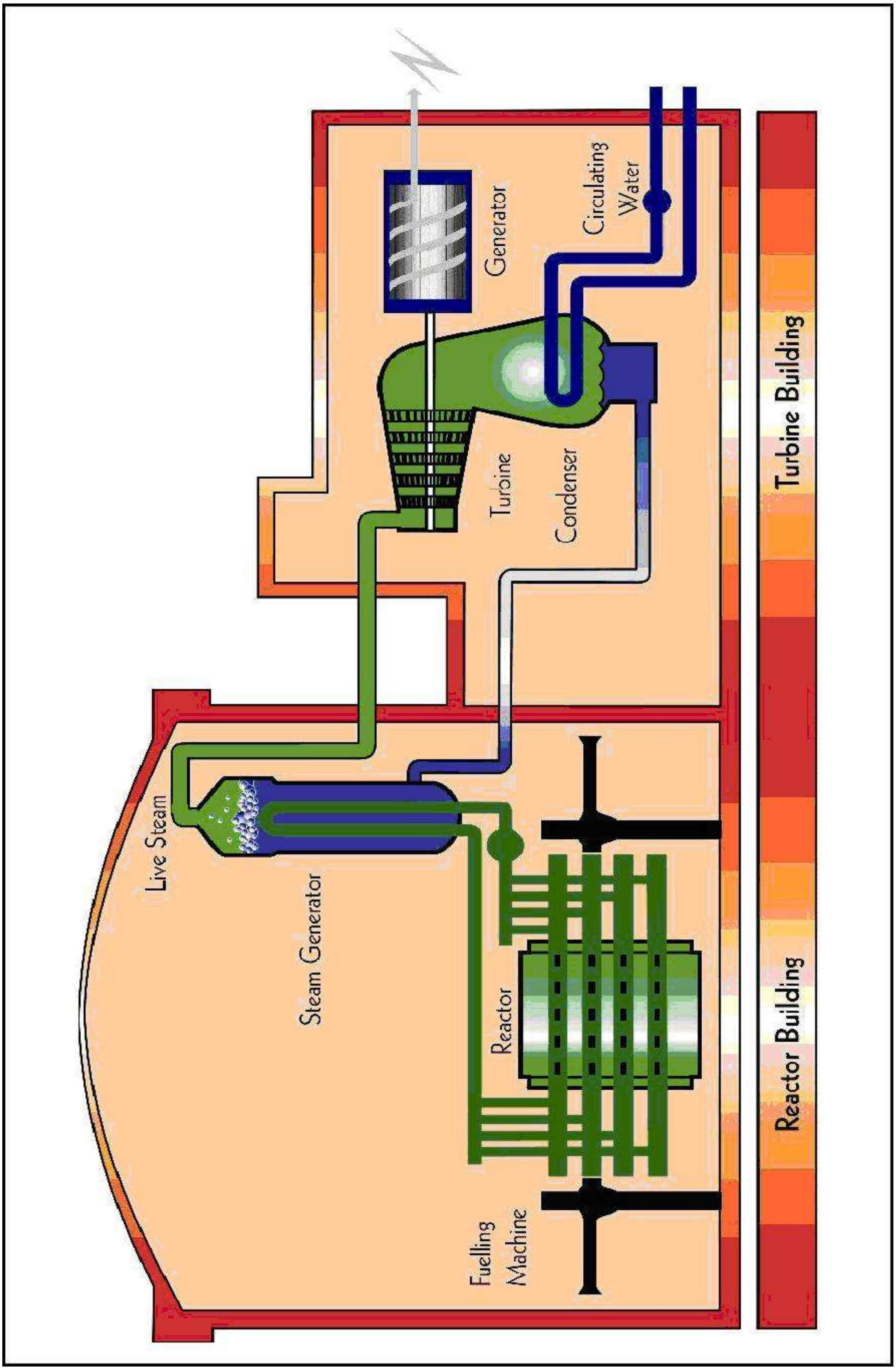
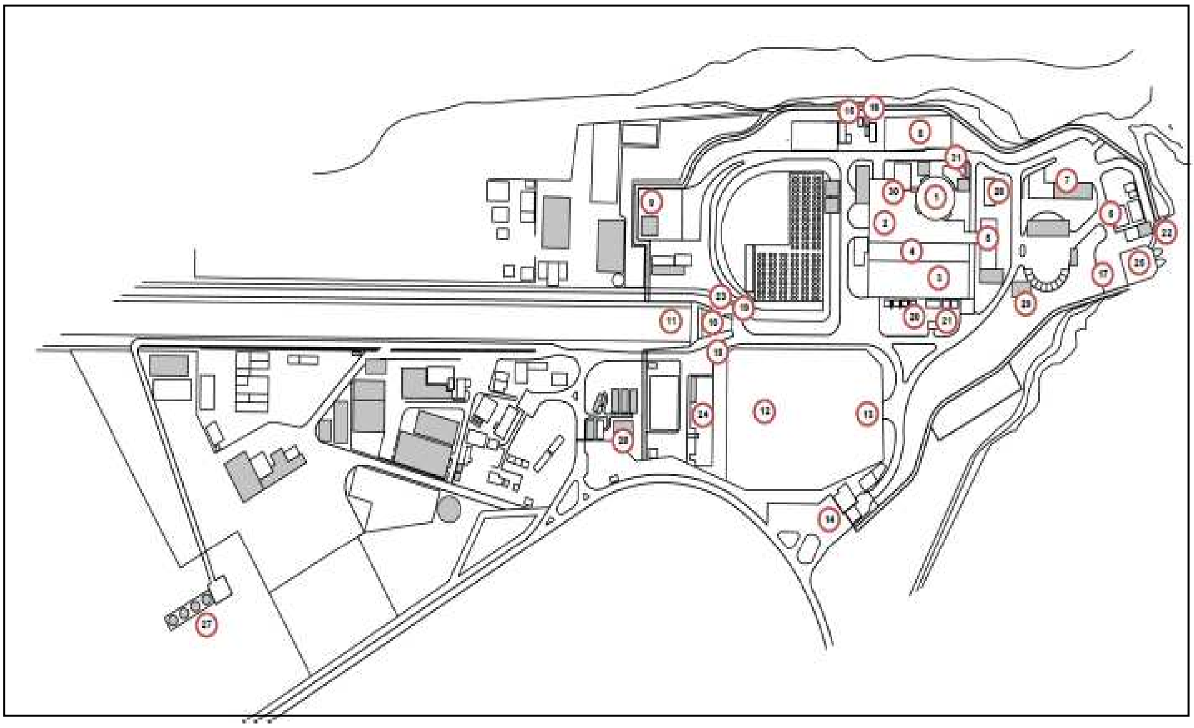


Figure III.2-2 - Embalse Nuclear Power Plant - Simplified Flow Diagram

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

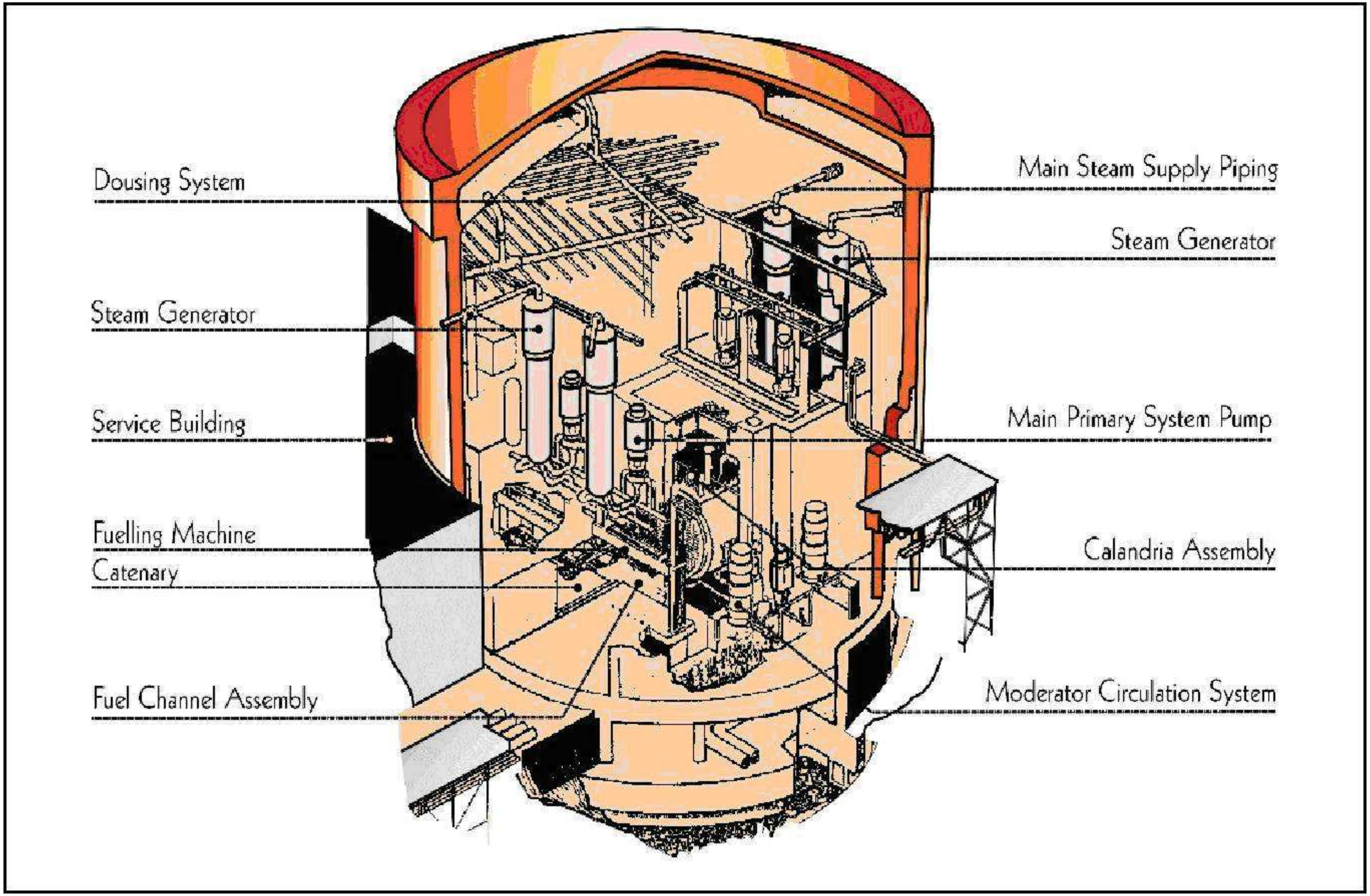
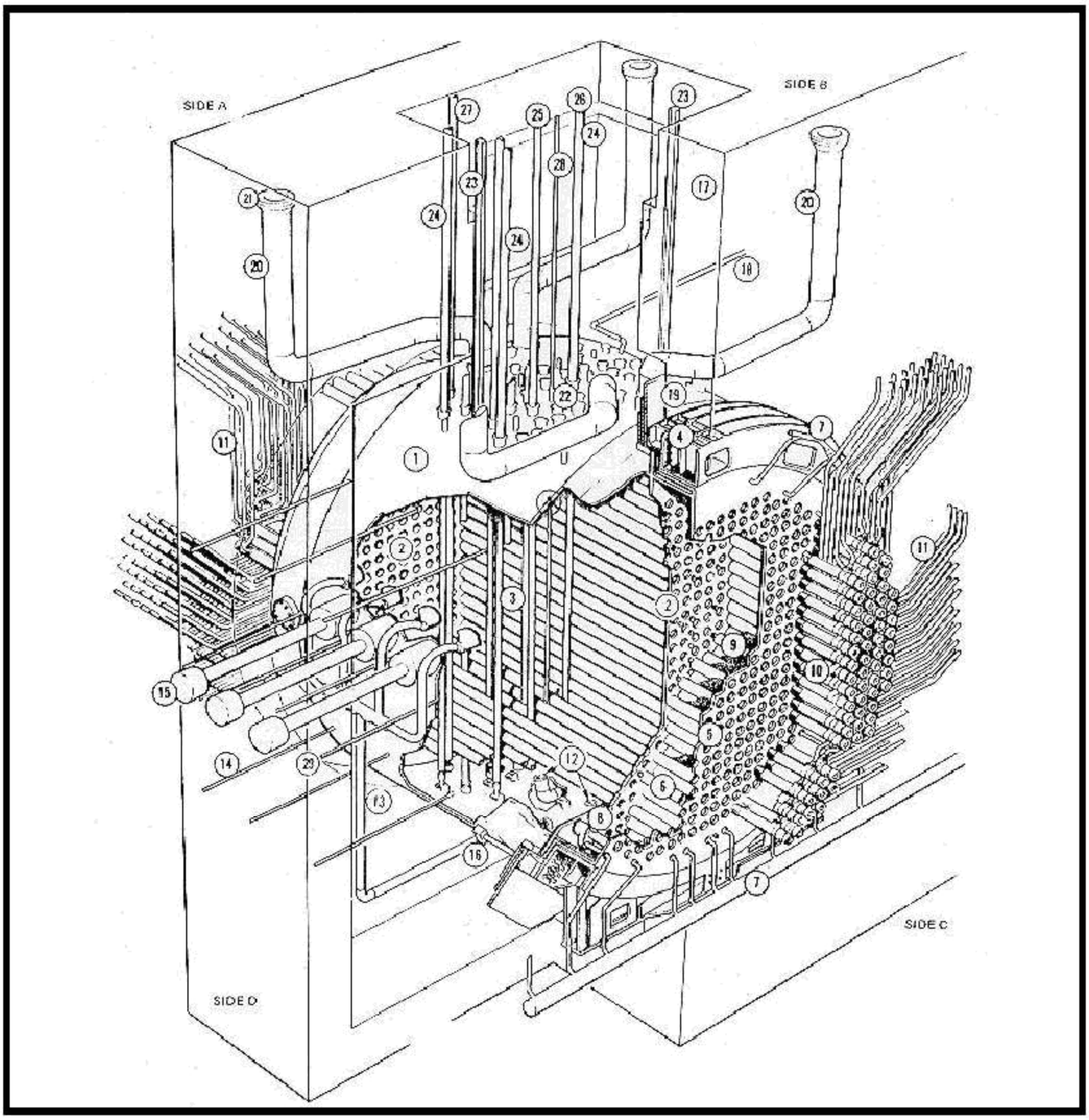


Figure III.2-4 - Embalse Nuclear Power Plant - Reactor Building

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

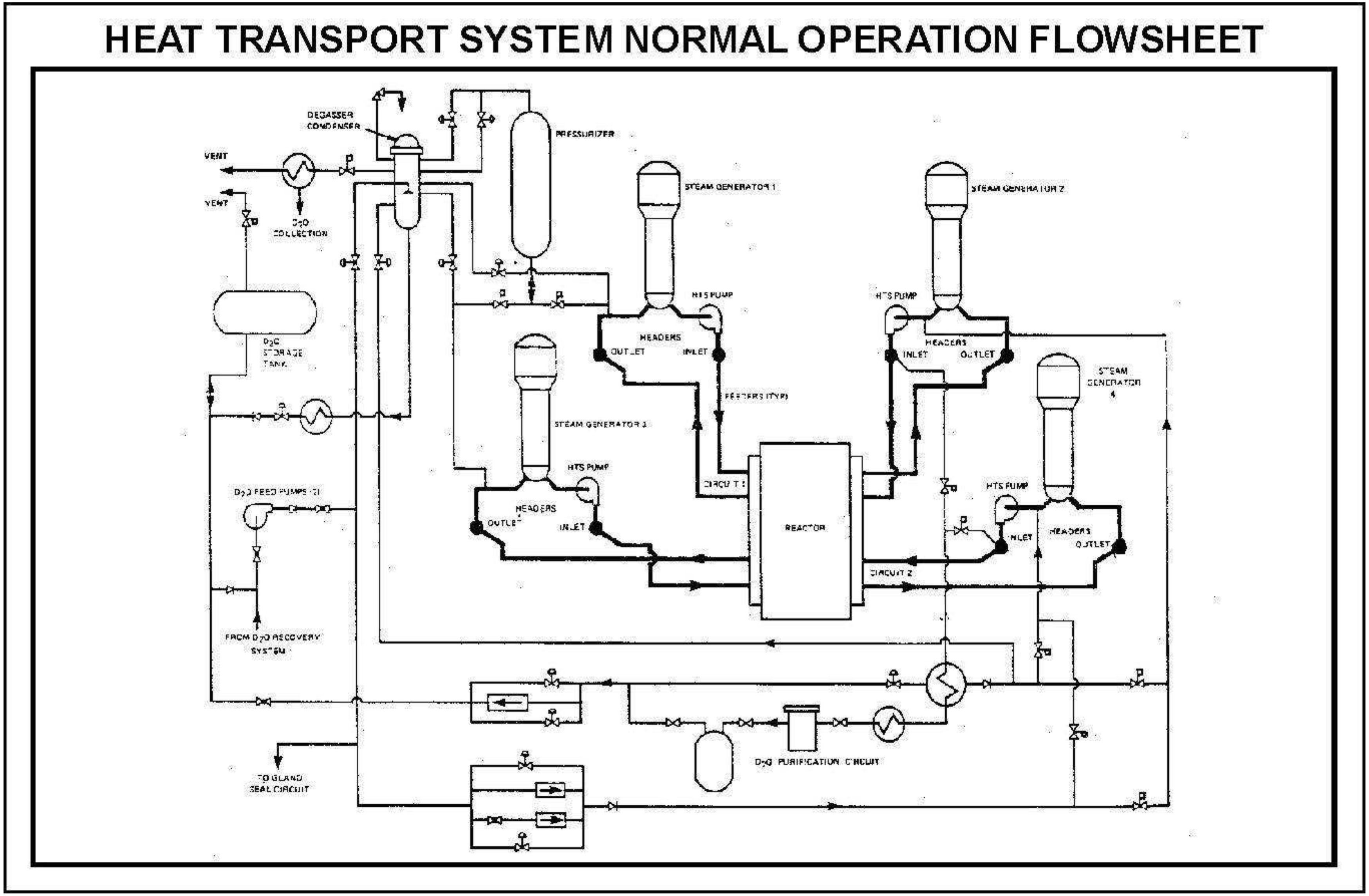


Figure III.2-6 - Embalse Nuclear Power Plant - Heat Transport System Normal Operation Flowsheet

SIMPLIFIED MODERATOR SYSTEM FLOW DIAGRAM

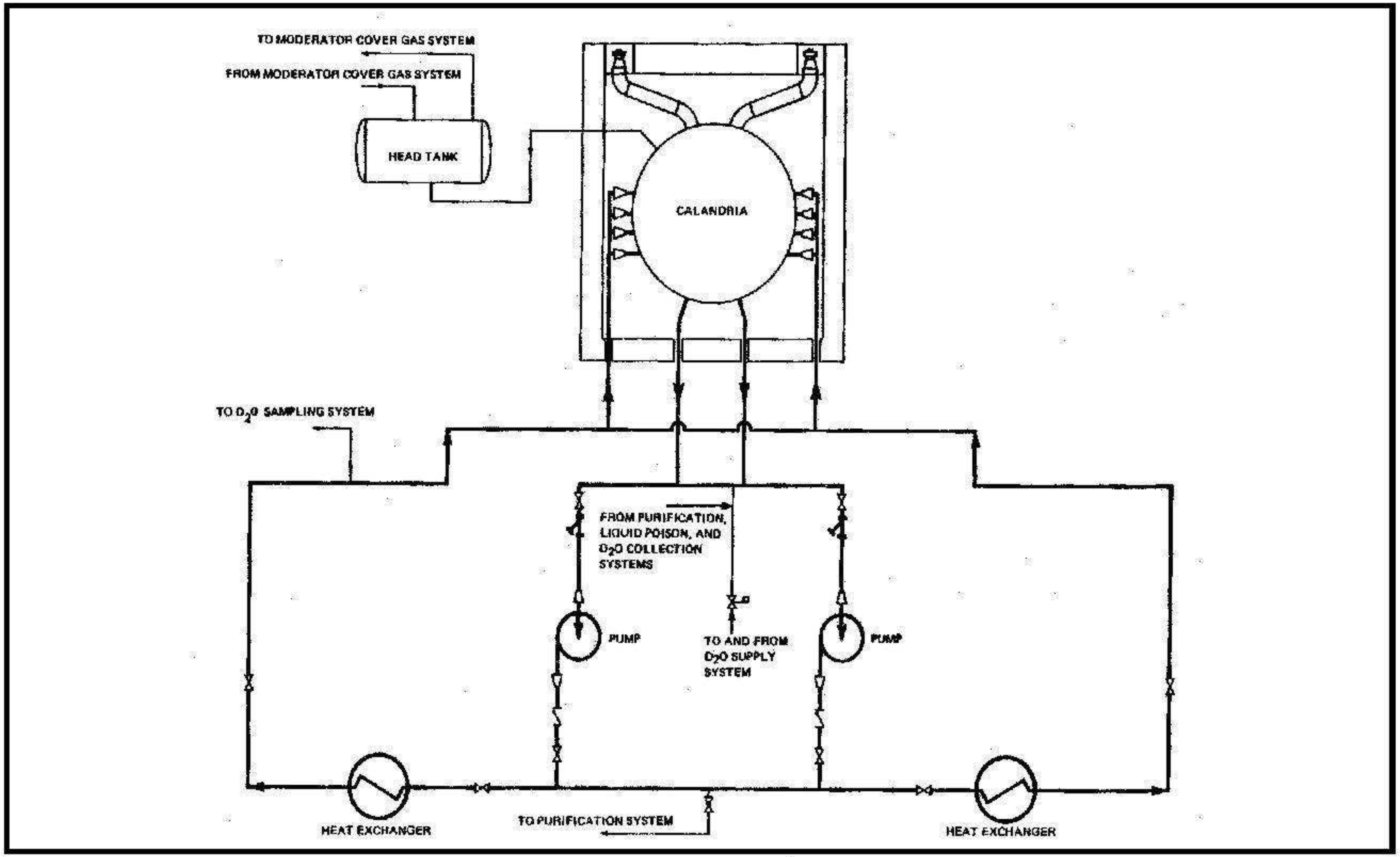


Figure III.2-7 - Embalse Nuclear Power Plant - Simplified Moderator System Flow Diagram

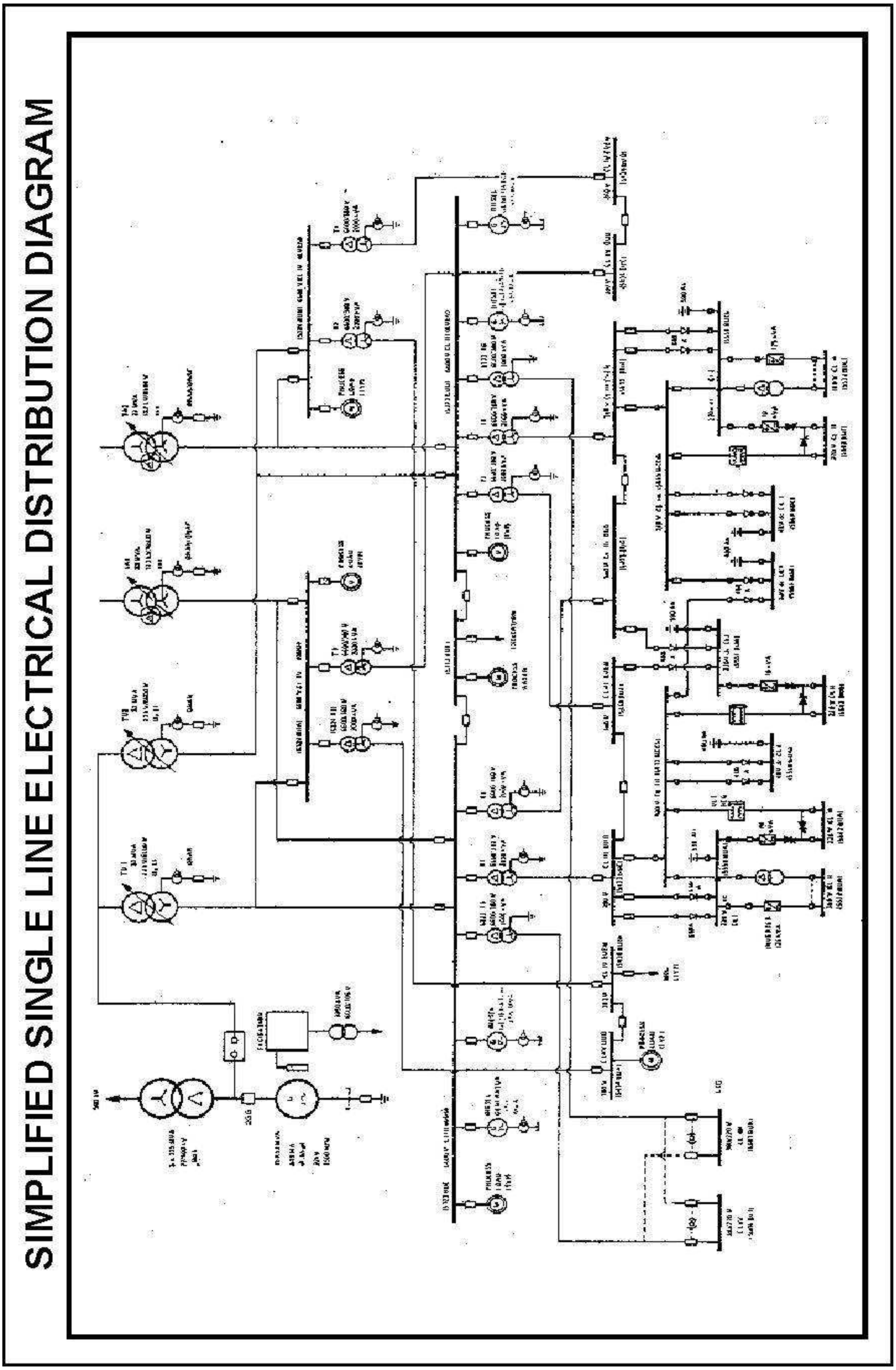


Figure III.2-8 - Embalse Nuclear Power Plant - Simplified Single Line Electrical Distribution Diagram

III.3. ATUCHA II NUCLEAR POWER PLANT

III.3.1. INTRODUCTION

The CNA II -a 745 MWe (2160 MWt) nuclear power plant– was designed by Siemens, with the participation of ENACE as architect engineer at the time the project began. CNA II construction license was issued in July 14th, 1981 and is currently under normal operation since 2016.

This PHWR made full use of KWU's experience in the light and heavy water reactor fields and the operating experience of CNA I, a plant that has shown excellent operating performance with high rates of availability and thereby furnished proof of its full operating reliability.

The pressure vessel type PHWR implemented in the 745 MWe PHWR is derived from CNA I and the 1300 MWe KWU standard PWR. Thus, the heavy water specific components such as moderator pumps, moderator coolers, moderator valves, coolant channel closures, refueling system, heavy water upgrading columns, etc. have an almost identical design compare to CNA I. Other components of the nuclear and conventional part of the plant, almost all main and auxiliary systems and the building layout are similar to KWU standard PWR design.

CNA II plant 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. The plant is located adjacent to the east side of CNA I (*Figure III.3-1*).

The reactor is pressure vessel type, fuelled with natural uranium and it is heavy water moderated and cooled. The moderator heat is used for preheating of steam generator feedwater and its temperature for reactivity control of the reactor. The reactor is on-load refueling with a single refueling machine arranged on top of the reactor pressure vessel.

Four redundant trains are installed for every safety system, thus enabling repair work on one train during plant operation. In every conceivable accident condition, the reactor plant can be kept "hot subcritical" or cooled down with the help of the high pressure residual heat removal system or with the emergency feeding of water on the secondary side of the steam generators as in PWR technology.

Figure III.3-2 shows schematically the main systems of CNA II.

III.3.2. Overall Plant Layout

The overall layout (*Figure III.3-3*) and arrangement of CNA II on the site is governed by the following basic considerations:

- Clear separation of the nuclear and conventional systems.
- Clear energy flow paths.
- Short piping and cable mans.
- Good transport conditions and access for construction, installation and operation.
- Physical separation of lines and ducts for redundant systems
- Consideration of radioactivity and the need for shielding
- Concentration of systems and components which are interconnected
- Assurance of internal and external safety

Buildings and structure arrangement of the CNA II are shown in (*Figure III.3-3*). As it can be seen from the site plan, the main buildings and structures of the plant are:

- Reactor building, incorporating containment structure and annulus
- Reactor auxiliary building with heavy water enrichment tower and vent stack
- Fuel store building, with storage areas for new and spent fuel assemblies
- Switchgear building, including the plant control room
- Turbine building

- Emergency power and chilled water supply building
- Main steam and feedwater valve compartment

CNA II has a clear physical separation between the nuclear and conventional sections of the plant. The reactor building, along with the annulus, reactor auxiliary building and the fuel storage building, constitutes the “controlled area” in which all systems assigned to the nuclear section are installed. In this way the radioactivity which arises is limited to defined regions. There is only one controlled access to the "controlled area".

All pressure retaining components of the nuclear steam supply system such as the reactor, the reactor coolant system, the moderator system and associated equipment are arranged inside the reactor building, which is enclosed by the inner spherical steel containment and the outer concrete shield. The containment structure is designed for the maximum pressure associated with the worst event which has to be taken into account.

A special ventilation system for the annulus ensures that even under accident conditions small radioactive leakages from the containment are retained by charcoal filters, thus preventing any radiation hazards to the environment. The systems necessary for on-load refueling are also housed in the containment structure.

In the lower part of the annulus between the containment sphere and the concrete shield various auxiliary and ancillary systems are accommodated, such as: residual heat removal system, safety injection system, heavy water storage system and components of the reactor cooling system.

The reactor auxiliary building adjoins the reactor building, and surrounds a part of it, thus allowing short connections to the equipment located in the reactor building annulus.

On the upper floors of the building there are active and inactive sanitary rooms, the laundry with ancillary rooms, the controlled access area ventilation system, the radiochemistry laboratory, the areas for radiation protection and the respiration apparatus room. From this area of the reactor auxiliary building there is an access to the reactor sphere via a personnel airlock.

The lower floors accommodate different auxiliary systems, such as: volume control system, heavy water purification and degassing system, heavy water treatment and enrichment system, boric acid and chemical control system and the gaseous, liquid and solid waste processing systems.

The fuel storage building is linked with the reactor building by the fuel transfer system. Personnel access is possible from the reactor auxiliary building.

Inside the building, there are four fuel storage pools, a manipulating pool, a small pool for the spent fuel shipping cask, a new fuel store and the necessary auxiliary equipment. The spent fuel assemblies are transferred from the reactor to the fuel storage pool with the aid of the fuel transport system, consisting of: refuelling machine, tilter, transfer tube, tilting device and manipulating bridge. The fresh fuel assemblies are supplied to the reactor in the reverse way.

The switchgear building has nine floors. They are used as follows:

- Cable ducts.
- Cable basement.
- High voltage switchgear.
- Cable race below D.C. systems.
- Battery, rectifiers, DC distribution boards.
- Cable race, instrumentation and control.
- Cabinets for instrumentation and control.
- Ventilation ducts, cable race below control room.
- Main control room, computer room, ventilation systems.
- Vent air system.

The switchgear building is attached to the shorter side of the reactor auxiliary building and houses in four similar sections the switchgears and electronic cublicles of the plant. Access to the switchgear building is from the staff facilities and office building via a personnel passageway. Access to the reactor auxiliary building is at the same level. The personnel passageway between staff facilities and office building/ turbine building allows passage between switchgear building and turbine building.

The off-site power transformer is located in front of the longitudinal side of the building facing the turbine building.

The turbine building is located adjacent to the reactor building with the turbine axis pointing in the direction of the reactor building. This gives maximum protection of the reactor building should the highly unlikely event of a turbine rotor burst occurs.

The building is of a two bay design. The main bay houses the turbine generator set and the feedwater heating equipment. The lower ancillary bay houses the feedwater tank, deaerators and feedwater pumps and other equipment associated with the water/steam cycle. All these compartments are free of radioactive media.

The main steam lines coming from the reactor building enter the turbine building along the shortest route leading to the area of the high pressure casing of the turbine, where the main steam flows through the steam strainers into the high pressure turbine. Vertical moisture separators are installed on both sides of the high pressure casing.

The basement of the turbine building is used mainly to accommodate pipes and cables. The heat exchangers for the low pressure feed heater drains, the closed circuit cooling water system and the associated pumps are also installed in the basement.

The generator busbars are routed from the generator to the generator transformers installed against the wall outside the turbine building, and to the high voltage plant service transformers.

The circulating water pipes enter and leave the turbine building on the same side.

The emergency power and chilled water supply building has two service floors. The building is further subdivided into four equal sections of similar construction which house redundant systems and equipment. This building is connected to the four redundant trais of the switchgear building via four separate pipe and cable ducts.

The diesel fuel storage tanks, pumps, secured component cooling heat exchangers, air recirculation system and the cable and pipe spreading rooms are installed on the lower floor.

The emergency power generators with their switchgear and the water-chilling units are installed on the upper floor. The diesel fuel day tanks and the start-up air supply system for each of the emergency power generators are installed on a gallery structure.

III.3.3. CNA II Main Systems

III.3.3.1. Reactor

The reactor (*Figure III.3-4*) is of the pressure vessel type, natural uranium fuelled and heavy water cooled and moderated. The total thermal power is 2160 MW.

The reactor core is approximately cylindrical in shape and consists of 451 natural uranium fuel assemblies located in the same number of coolant channels. Each fuel assembly consists of 37 fuel rods arranged in three concentric circles, the rod supporting plate, the spacers for lining up the fuel rods, and the linkage with a coupling for connection to the filler body. Each fuel rod consists of a stack of uranium dioxide pellets enclosed by a thin walled zircaloy 4 canning tube, which is both gas and pressure tight. Each fuel assembly, together with the filler body and the closure plug, forms the fuel bundle column. The coolant channels are arranged vertically in a triangular lattice within the moderator tank. The fuel bundle columns can be removed from the coolant channels during reactor operation by the refueling machine. The filler bodies serve to reduce the volume of the coolant in the reactor coolant system.

The heat generated in the fuel assemblies is transferred to the reactor coolant, which flows through the coolant channels and transports the heat to the steam generators.

The coolant channels are surrounded by the moderator, which is enclosed in the moderator tank. For reactivity reasons, the moderator is maintained at a lower temperature than the reactor coolant. This is accomplished by the moderator system, which extracts the moderator from the core, cools it down in the moderator coolers, and feeds it back into the core. The heat removed from the moderator is used for pre-heating the feed-water. This is one of the reasons for the high net efficiency (approx. 32%) of the NPP.

The reactor coolant system and the moderator system are connected by the pressure equalization openings of the moderator tank closure head. Therefore, the pressure differences in the core are comparatively small, which results in thin walls for the reactor pressure vessel internals. This allows a very high burn-up to be attained. Furthermore, the connection between the reactor coolant system and the moderator system permits the use of common auxiliary systems to maintain the necessary water quality. The number of auxiliary systems can therefore be reduced to a minimum.

For control of the reactivity, and thus of the power output of the reactor, various methods are applied. The reactor contains nine "blacks" (absorbers made of hafnium) and nine "grey" (steel) control elements arranged in 3 groups. The control elements are used to control the reactivity and the power distribution, to compensate the build-up of xenon poisoning following a reactor power reduction, to provide damping of azimuthal xenon oscillations, and to shut down the reactor. The reactivity value of all control elements is sufficient to shut the reactor.

In addition to the control elements, reactivity control is provided by the boric acid dosing system. The injection or extraction of boric acid serves to compensate slow reactivity changes due to the burnup during the first period of operation and to maintain the reactor in a safe subcritical condition at zero power. Extraction of the boric acid is performed by anion exchangers.

Additionally, a boron injection system, as a second independent shutdown system is provided, which injects boric acid into the moderator.

The reactivity can, in addition to these reactivity control systems, also be controlled by varying the moderator temperature within a certain range, which is advantageous for some operating modes.

The reactor pressure vessel (RPV) constitutes the pressure boundary of the reactor core and encloses the core components and the reactor pressure vessel internals. The RPV consists of the lower part, the closure head and the studs and nuts which connect both sections. The connection is made leak-tight by means of a welded lip seal.

The lower part of the RPV consists of the hemispherical bottom section, two shell courses and a shell flange which carries the coolant inlet and outlet nozzles and the support pads located between them. The reactor coolant inlet and outlet nozzles are arranged on one plane; there are no penetrations or pipe connections below this plane. The reactor core is housed below the plane of the inlet and outlet nozzles.

The closure head consists of a flange and a dome plate connected by a circumferential weld. The closure head dome carries the nozzles for coolant channels, moderator pipes, and control element drives and for in-core instrumentation. The nozzles are screwed into holes in the closure head dome and sealed by an overlay weld.

Most of the RPV internals form the structure of the reactor core. The moderator tank accommodates all core components, separates the moderator from the coolant and, in conjunction with the reactor pressure vessel, forms the annulus for the in-flowing coolant. The bottom of the moderator tank serves as the lower fixing level for the coolant channels and the control element guide tubes. The moderator tank shell serves as thermal shielding.

The moderator tank closure head form the upper plenum for the reactor coolant leaving the coolant channel slots. The closure head and the moderator tank jointly form a unit which keeps the coolant channels and the control element guide tubes in position firmly and without displacement during all operating modes, as well as during postulated accidents. The moderator tank and its closure head are suspended from the flange of the reactor pressure vessel and are held tightly in position by the pressure vessel closure head.

The coolant channels consist of vertically arranged tubes which contain the fuel bundle columns, direct the reactor coolant flow and separate the reactor coolant from the surrounding moderator.

The reactor coolant flows inside the coolant channels in an upward direction. After 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 refueling machine during reactor operation in order to exchange the fuel bundle column located inside the coolant channel.

The moderator piping serves for supply, distribution and extraction of the moderator inside the moderator tank. The moderator piping essentially encompasses four down-comers, the sparger ring on the moderator tank bottom, and the suction boxes with nozzles in the moderator tank closure head.

The moderator flows downwards through the down-comers to the sparger ring, where it is distributed at the moderator tank bottom. After rising and heat-up in the moderator tank, the moderator flows to the suction boxes and leaves the moderator tank through two nozzles.

The filler pieces are provided in the reactor pressure vessel in order to displace the heavy water and thus reduce the heavy water inventory rewired. The upper filler pieces are adapted to fit the reactor pressure vessel closure head. The lower filler pieces are divided into several interlocking rings and adapted to fit the bottom head of the pressure vessel.

III.3.3.2. Reactor coolant system and moderator system

The reactor coolant system and the moderator system (*Figure III.3-5*) remove the 2160 MW thermal power generated in the reactor core and the approximately 15.8 MW heat generated by the reactor coolant pumps to the feedwater/steam cycle.

The reactor coolant system transfers the heat generated in the reactor core via the steam generators to the turbine. The system is structured similar to that of a pressurized light water reactor and consists of two identical loops, each comprising a steam generator, a reactor coolant pump and the interconnecting piping, as well as one common pressurizer and pressurizer relief system.

The heat is carried by the reactor coolant, which flows from the reactor pressure vessel to the steam generators, where it is cooled down and then pumped back to the reactor pressure vessel by the reactor coolant pumps.

The pressurizer system is connected to one reactor coolant loop and basically comprises the pressurizer with the electric heaters, the surgeline, the spray lines and valves, and the safety valves.

The function of the pressurizer system is to maintain the appropriate pressure in the reactor coolant system in order to prevent boiling of the coolant under all operating conditions (principle of the pressurized water reactor), and to avoid or limit the pressure variations caused by volume fluctuations during load changes. The pressurizer is partly filled with saturated water and partly with steam. If the pressure drops, water is evaporated by switching on the electric heaters, raising the pressure to its set point. In the event of a pressure rise, steam is condensed by spraying water into the steam space.

Besides pressure control by sprays in the pressurizer, protection against overpressure in the reactor coolant system is provided in accordance with international codes for pressure vessels and systems. Protection is afforded by independent, self-actuating safety valves.

When the safety valves open, the steam discharged from the pressurizer is directed into the pressurizer relief system, where it is condensed to water.

The steam generators transfer heat from the reactor coolant on the primary side to the feedwater/ steam cycle on the secondary side. The transferred heat raises the feedwater temperature and generates the saturated steam which drives the turbine generator unit. The steam generator constitutes the barrier between the radioactive reactor coolant and the non-radioactive feedwater/steam cycle, preventing the carry over of radioactive matter.

The steam generator is a vertical U-tube heat exchanger with natural circulation of the feedwater on the secondary side. The primary side of the steam generator consists of the channel head (primary plenum) and of the heating tube bundle. On the secondary side, the feedwater enters through a nozzle located in the steam dome and is distributed by a feedwater ring manifold in the annulus, (down-comer) formed by the secondary side shell and the tube bundle wrapper. The feedwater flows downwards to the tube sheet and enters the tube bundle region (riser). The feedwater is heated and partly evaporated around the U-tubes in this region. The generated steam-water mixture leaves the riser region and flows through cyclones and steam driers arranged in the steam dome. The dried steam is discharged through the main steam outlet nozzle, the separated water flows downwards into the down-comer where it is mixed with the incoming feed-water.

The moderator system consists of four identical loops operating in parallel. Each loop comprises a moderator cooler, a moderator pump, and the interconnecting piping with valves.

The moderator system performs various functions depending on the operating mode of the reactor.

During normal operation, the moderator system maintains the moderator at a lower temperature than that of the reactor coolant. The moderator leaves the top of the moderator tank, flows to the moderator pumps, is pumped to the moderator coolers and flows back to the bottom of the moderator tank. The heat transferred in the moderator coolers is used for preheating the feedwater.

For residual heat removal, the moderator system is switched over to the residual heat removal position by means of the moderator valves. Under this mode of operation, the moderator is extracted from bottom of the moderator tank by the moderator pumps and fed into the cold legs of the reactor coolant loops, and also directly into the reactor coolant inlet annulus of the reactor pressure vessel, via the moderator coolers. The moderator system forms the first link of the residual heat removal chain. The residual heat is transferred from the moderator system to the residual heat removal system and then to the service cooling water system.

During emergency core cooling, the moderator serves as a high pressure core reflooding and cooling system. The emergency core cooling position is similar to that of the residual heat removal, but additionally, water is injected into the hot legs of the reactor coolant loops and into the upper plenum of the reactor pressure vessel. The residual heat removal chain connected to the moderator coolers during emergency core cooling is the same as during residual heat removal.

All systems of the residual heat removal chain are of a consistent “four loop” design. The residual heat removal system acts as a barrier between the active moderator and the service cooling water and prevents the escape of radioactivity into the service cooling water in the event of leakages in the moderator coolers.

III.3.3.3. Refueling system

The natural uranium reactor makes it possible and desirable, with a view to obtaining a high burnup, to shuffle and replace the fuel assemblies during power plant operation. The refueling procedure is carried out by a single refuelling machine. The fuel assembly transport system is located in the reactor building and in the fuel pool building. The main items of the fuel transport system are: refuelling machine, tilter with supporting structure, fuel transfer tube, fuel pool, and the corresponding auxiliary systems and maintenance installations. The refueling procedure is fully automated and monitored from the control room.

The refueling machine is moved from a maintenance position in the refueling machine maintenance room, by remote control, to a previously selected coolant channel position in the reactor vessel. in which the machine is centered. The seat-on seal is pressed on to the coolant channel closure head by the dead weight of the refueling machine to form a water-tight seal between the machine and the coolant channel. A pressure equalization takes place between the refueling machine and the reactor before opening the isolation valve of the refueling machine and opening the coolant channel closure. Following this, the fuel bundle column is withdrawn into an empty position in the refueling machine magazine. The magazine is then rotated in such a way that a fuel bundle column with a partially burnt up fuel assembly or with a new fuel assembly is positioned above the open coolant channel. This fuel bundle column is lowered into the coolant channel position and the coolant channel closure is locked again. After closing the isolation valve of the refueling machine a check for leak tight closure is performed. Then the refueling machine is removed from the reactor pressure vessel and is positioned above the vertically arranged tilter. The tilter has the following functions in the sequence indicated:

- Take-over of the fuel bundle column with the spent fuel assembly.
- Removal of the decay heat by cooling with D_2O.
- Drying and cooling the spent fuel assembly with gas.
- Flooding and cooling of the tilter with H_2O.
- Tilting to the horizontal position and connecting with the fuel transfer tube.
- Transfer of the fuel assembly into the fuel transfer tube.

When a new fuel bundle column is transported from the fuel pool building into the tilter via the transfer tube, and later from there into the refueling machine, the process of cooling and change of cooling medium takes place in the reverse order.

The fuel transfer tube connects the reactor building to the transfer pool in the fuel pool building. The isolation valves in the transfer tube permit inward and outward transfer without degrading the containment isolation function.

The tilting device takes the fuel bundle column from the fuel transfer tube and swivels it from the horizontal into the vertical position.

A silver-clad seal ring with good material flow properties is used as sealing material. In this established design, the coolant channel seals are almost perfectly tight.

III.3.3.4. Reactor auxiliary and ancillary systems

The auxiliary systems are basically organized in the same way as the auxiliary systems in PWR plants and work together with the reactor coolant system and moderator system to ensure the specified chemical conditions of the coolant and moderator. The systems containing heavy water are strictly separated from the systems containing light water in order to avoid downgrading the heavy water. The main tasks of the auxiliary systems are:

- Storage of heavy water.
- Volume control, seal water supply.
- Treatment and upgrading of heavy water.
- Boric acid dosing and chemical feeding into the primary circuit.
- Fast boron injection.
- Nuclear component cooling.
- Fuel pool cooling.
- Supply of refueling machine with auxiliary fluids.
- Compensation of leakages.
- Removal of decay heat from the core, emergency core cooling.

The auxiliary and ancillary systems are located mainly in the auxiliary building and partly in the annulus of the reactor building.

III.3.3.5. Main control room and emergency control room

The main control room of the nuclear power plant contains the operating and information equipment for the control and monitoring of the plant systems. This means that manual control, setpoint adjustment and monitoring of the reactor, important reactor auxiliaries, the feedwater/steam cycle, the turbine, the generator and the auxiliary power equipment are controlled from the main control room.

The main control room is situated on the top floor of the switchgear building above the electronic equipment rooms. For security reasons, it shall only be entered through monitored entrances.

In the event of unhabitability of the main control room during power operation, the operators can be directed to the emergency control room (ECR). The ERC contains the equipment necessary to bring the plant to a safe shutdown, maintain it in a safe condition and monitor the essential parameters of the plant.

III.3.3.6. Instrumentation and control systems

The instrumentation and control equipment includes the measurement, control, protection and monitoring system.

The control room is connected to the drives and signal transmitters in the plant through the automation and plant interface equipment. Plant conditions and operational transients are transmitted via analog and binary signal transmitters to both the operator in the control room and to the automatic equipment. The command signals to the drives in the plant are transmitted via the control interface as required for maintaining proper operation.

Measured-value and status signals from the entire plant are continuously monitored by means of a process computer. The computer provides the data for trend logging and fault analysis and transmits the information to the operating personnel in the control room via data display terminals and teleprinters.

Automatic functional group controls are provided to minimize the operating errors and to obtain a higher degree of automation.

III.3.3.7. Electric power system

The CNA II has two physically independent grid connections. One of then is the 500 kV grid and the other is the 132 kV grid.

There is additionally, auxiliary power supply from the generator in case of grid disturbance after load rejection. Only in the case of a common outage of all three power supply possibilities, the emergency power system with the diesel generators will be required. Definite loads – mainly of the control and instrumentation system – are power supplied by rectifiers and converters or by means of batteries with direct current.

The generator feeds into the 500 kV network via one generator transformer and supplies the plant auxiliary service requirements by means of two auxiliary transformers.

The four secondary windings of the auxiliary transformers or the two secondary windings of the offsite system transformer feed into four separate medium high voltage bus section (each two 6.6 kV and 13.2 kV), to which the large auxiliary loads and the transformers for the low voltage switchgears are connected.

If the plant service power system fails, a certain equipment (pumps, etc.) is needed to remove residual heat and to run the plant into safe conditions. This equipment must remain in operation or has to be put in operation and must therefore be supplied with emergency power. Here a distinction must be made between two groups with regard to safety related requirements: loads allowing a voltage interruption while the diesel run up and loads which must remain in operation without interruption or which must be put into operation immediately should the normal plant service system fails.

The diesel emergency power system is like the other redundant safety equipment, divided into four separate trains. Under normal operating conditions, the auxiliary switchgears of the normal power system feed the emergency power system. To avoid loss of power in case the auxiliary power system fails, each of the four trains in the emergency power system is equipped with a quick-starting diesel set.

The CNA II electric system may be divided into two main subsystems: the offsite power system and the onsite power system.

The offsite power system is constituted by the 500 kV transmission line, which is linked to the substations Rosario Oeste (113 km), Colonia Elía (160 km) and Ezeiza (67 km), and the 132 kV transmission line, source that is connected to the Zárate substation (23 km) and, in addition, to the 220 kV switchyard at the power plant Atucha via a 150 MVA coupling transformer.

With the generator load-breaker in the “off” position the plant service power for “start up” and “shut down” of the power plant can be drawn from the 500 kV grid. Upon simultaneous failure of the main grid and the turbine generator set the 132 kV grid provides power for shut down operation of the plant down to the “hot-subcritical” condition.

The onsite power system consists of two subsystems: the auxiliary power system (“normal system”) and the emergency power system.

The auxiliary power system provides power for the loads of the nuclear power plant which are necessary during normal operation, start-up and shut-down operations. It is subdivided into four trains which are supplied by the 13.2 kV and 6.6 kV windings of the two unit auxiliary transformers. The transformers are fed in via single phase totally enclosed leads either from the min generator or from the grid via the external generator transformer. For shut-down operation or after loss of the normal power supply grid and the generator, it may be fed by the offsite power supply. The offsite power supply is available via automatic changeover.

The A.C. emergency power system provides the power required for safe shut-down of the reactor to maintain it in the shut-down condition, for removal of residual heat and to prevent release of radioactivity during normal operation and accident conditions resulting from system faults, and for some loads important for plant availability. It is subdivided, according to the safety system to be power supplied, into four redundant independent trains, each capable of supplying 50% of the power required to perform the safety function.

Normally the A.C. emergency power supply system is connected with the two 6.6 kV buses of the plant auxiliary power system by means of two circuit breakers, connected in series, for each train. Therefore, it can be fed via the plant auxiliary power system or the offsite system. In case of loss of the plant auxiliary power it is fed by the diesel generator emergency power system supply.

The stand by diesel generator emergency power system is provided for safety related loads. Each diesel generator set is assigned to one train, each with 50% capacity. Each emergency diesel generator is constituted by different main and auxiliary subsystems, such as the compressed air, the fuel supply equipment, the lubrication system, etc.

In the event of the power plant and the high voltage system failing at the same time, an emergency power supply can be obtained from diesel sets to permit proper shutdown of the plant. On failure of the auxiliary voltage, a period of 20 to 30 seconds elapses before the emergency diesel sets can take load. However, certain loads, such as the reactor protection system, measuring systems, and other protective systems must remain operative at all times. These are therefore fed directly from 220 V or 24 V D.C. batteries or from static converters fed from the 220 V breakers.

III.3.3.7. Safety philosophy and safety systems

The safety philosophy, on which the design is based, fulfills, in all conceivable plant conditions, the following basic requirements:

- The reactor can be safely shut down and kept shut down over prolonged periods.
- The decay heat can be reliably removed.
- Any release of radioactivity is within the limits laid down by the radiation protection regulations.

In order to meet these requirements, safety measures against damage to the systems or components are provided. The safety measures can be classified under three safety levels according to the possible plant conditions:

- Components and systems necessary for normal operation (including startup, part load and full load operation, load changes, shutdown) are of such design as to preclude failure. The safety measures provided are:
  - Conservative and careful design.
  - Stringent quality assurance and control.
  - Regular examinations and inspections.
- According to general engineering experience, it must be considered that systems and components can fail during their service life despite adequately high quality. It is assumed that operational disturbances (e.g. reactor coolant pump failure, load rejection) can occur. In order to prevent the faults and operational disturbances and to mitigate their consequences the following safety measures are provided:
  - Inherently safe operational characteristics.
  - Alarm annunciation.
  - Reactor protection limitation.
- Despite the safety measures of the first and second safety levels, theoretically assumed accidents are postulated. In order to counter these accidents and to mitigate their consequences, active safety systems are provided. The design of the safety systems is based on the assumption that parts of the safety systems (sub-systems) can fail simultaneously with the accident. As a consequence, the safety systems are of redundant design. This multiple-train design is reflected not only in the redundancy of the equipment, but also in the consistent physical segregation of the subsystems. This ensures that a sub-system failure (random failure), postulated in addition to the accident, remains restricted to one subsystem and does not affect the others.

According to the design principles of the safety systems all engineered safety features necessary to control accidents are built as four identical sub-systems. Two of these are sufficient for the control of the accident. Thus, functional availability is assured even when one subsystem is being inspected or repaired and a single failure occurs simultaneously in another subsystem. Each subsystem also comprises the associated power supply and the necessary auxiliary equipment.

The basic safety systems provided are:

- Fast reactor shutdown system.
- Emergency core cooling system.
- Containment system.
- Emergency electric power system.

In order to protect the environment against the release of radioactivity, the following radioactivity barriers are provided as passive safety measures:

- The fuel matrix of the uranium dioxide pellets.
- The seal-welded cladding tubes enclosing the fuel.
- The closed and seal-welded reactor coolant system and moderator system.
- The full pressure gas tight steel containment structure.
- The concrete secondary shield.

The components of the radioactivity barriers act according to their mechanical properties, without auxiliary energy. In case of damage to one of these barriers, the next one will act and thus retain the radioactivity.

The accidents considered in the plant design are the plant internal and external accidents. The internal accidents are, above all, LOCA, with the whole spectrum of pipe ruptures including the break of the largest connection pipe to the reactor coolant loops or to the moderator system. The external accidents considered are aeroplane crash, explosion pressure wave, floods, tornadoes, etc.

In order to meet the safety requirements even during the considered internal and external accidents, the following design principles were established:

- Multiplicity of safety features.
- Redundancy of the safety systems and of their auxiliary system.
- Diversity of important parts of the reactor protection system.
- Physical separation and/or protection by concrete walls of the redundant sub-systems.
- Protection of the safety systems against external accidents.
- Periodic testing of the safety systems.

The task of the safety systems is to prevent any damage to the radioactivity barriers during operational malfunctions and during accidents in order to fulfil the safety philosophy requirements.

The fast reactor shutdown safety system consists of two separate subsystems: the shutdown control rod system (first independent shutdown system) and the boron injection system (second independent shutdown system). The emergency core cooling safety system consists of the following basic subsystems: the moderator system, the residual heat removal system, the service cooling water system for the secured plant, the nuclear component cooling system and the safety injection system.

The containment safety system consists of several basic subsystems: the concrete containment, the steel containment, the containment isolation system and the reactor building annulus air extraction system (*Figure III.3-6*).

The emergency electric power system is not a safety system but a safety related one due the nature of the loads it feeds.

The general plant layout features a compact, controlled area with short pipe and cable connections, physical separation of redundant cable and piping trains, minimum exposure of the operating personnel during maintenance and repair work, availability of sufficient shielding and radiation protection measures.

III.3.3.8. Technical Data of Atucha II Nuclear Power Plant

Some of the main technical data are detailed in what follows:

Overall Plant Data	
Reactor type	Pressurised heavy water (PHWR)
Bulk nominal electric power	745 MW
Net nominal electric power	692 MW
Net efficiency	32%
Number of steam generators	2
Number of reactor coolant pumps	2
Number of moderator coolers and moderator pumps	4
Reactor Core Data	
Material of fuel	Natural uranium
Total thermal power	2160 MW
Shape of fuel assembly	37 – rod cluster
Number of coolant channels or fuel assemblies	451
Cladding material	Zircaloy 4
Active core length	5300 mm
Refuelling	On load
Fuel burnup at equilibrium	7500 MWd/MgU
Number/material of control rod elements	9/hafnium and 9/steel
Thermal and Hydraulic Data	
Number of coolant circuits	2
Number of moderator circuits	4
Coolant and moderator	Heavy water
Pressure at reactor vessel outlet	115 bar
Coolant temperature at reactor pressure vessel outlet	312°C
Coolant temperature rise though the core	34°C
Moderator temperature normal/maximum	170°/220°C
Total coolant circulation flow	37080 Ton/h
Total moderator circulation flow	3200 Ton/h
Steam pressure at steam generator outlet	56 bar
Steam temperature	271°C
Total steam flow	3445 Ton/h
Steel Containment Data	
Diameter	56 m
Wall thickness	30 mm
Design pressure	5.3 bar
Design leak rate	0.25 vol %/day

SITE LOCATION

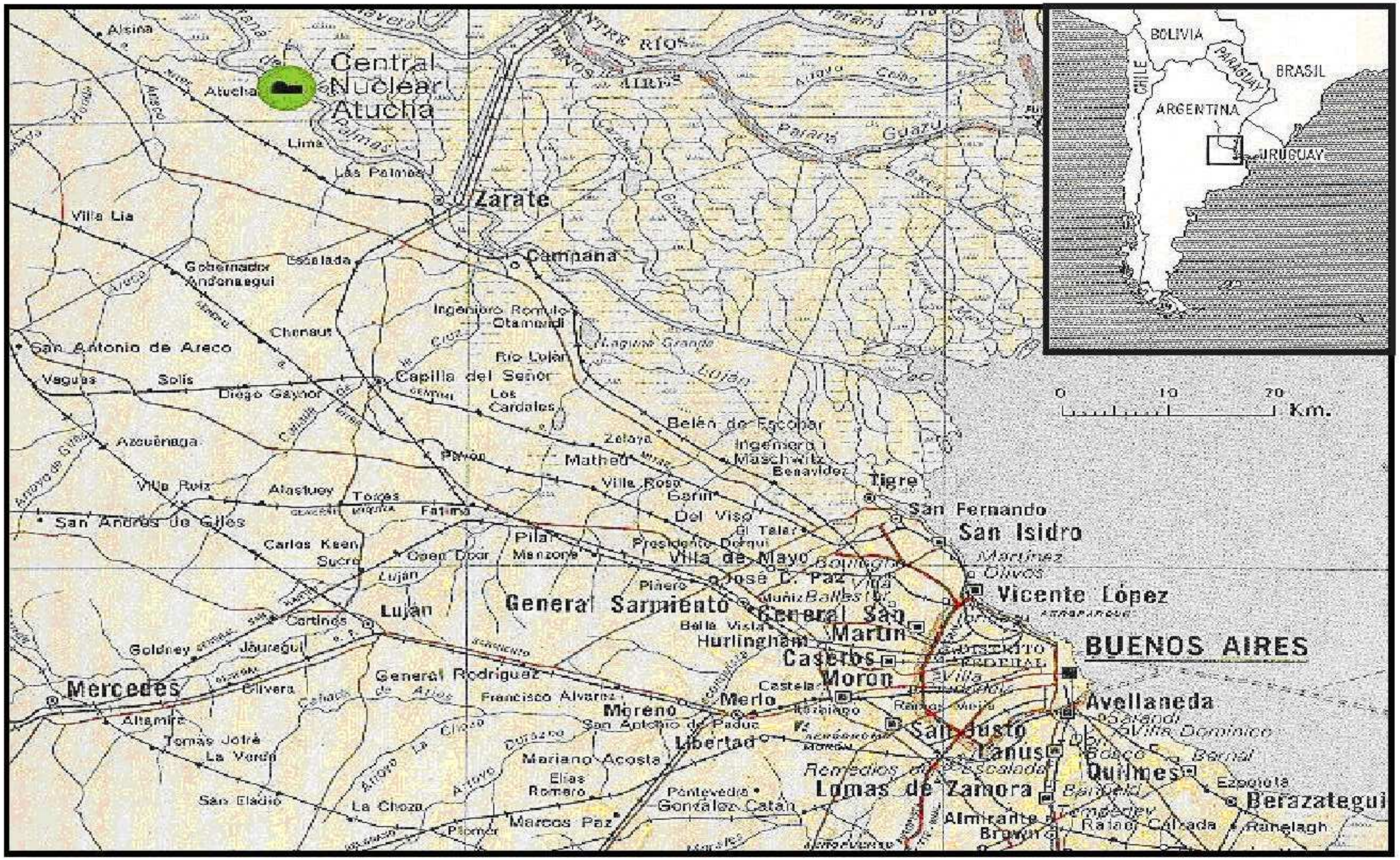


Figure III.3-1 Atucha II Nuclear Power Plant - Geographic Location

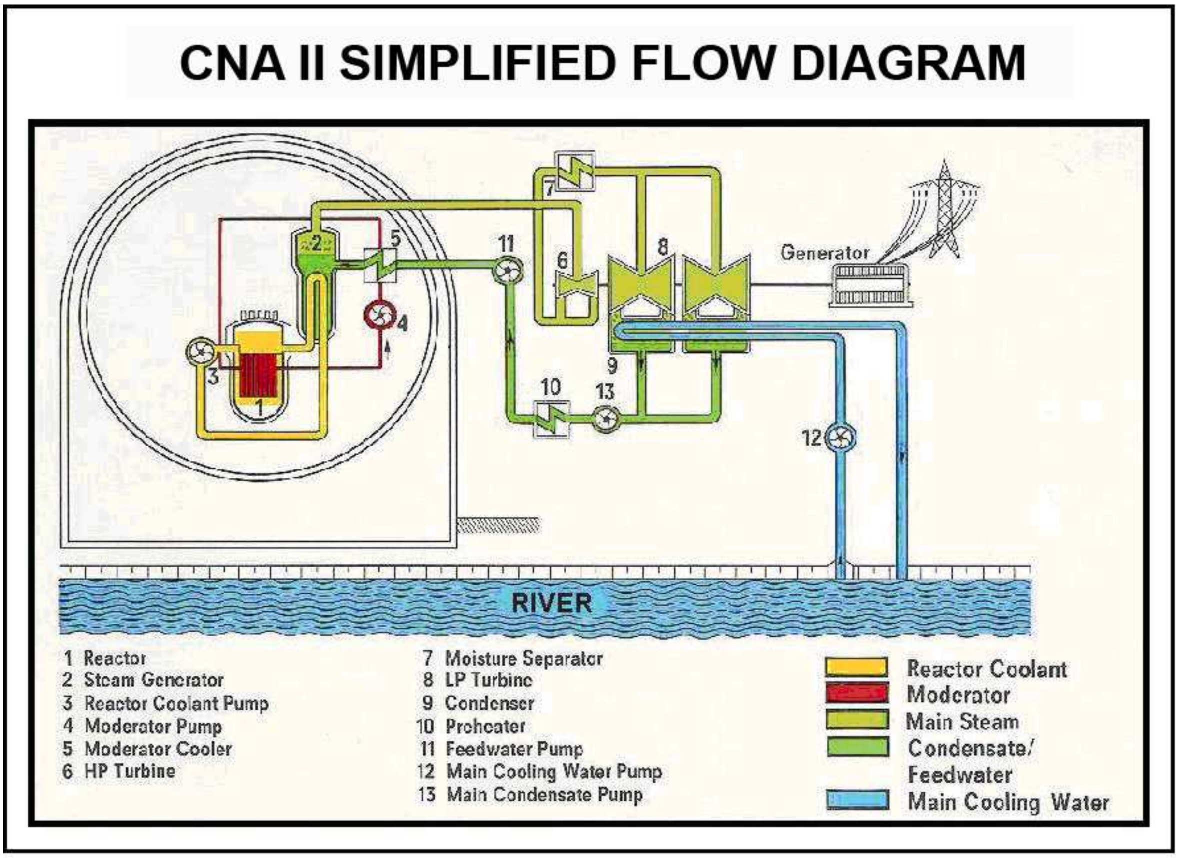


Figure III.3 - 2 Atucha II Nuclear Power Plant - Main Systems

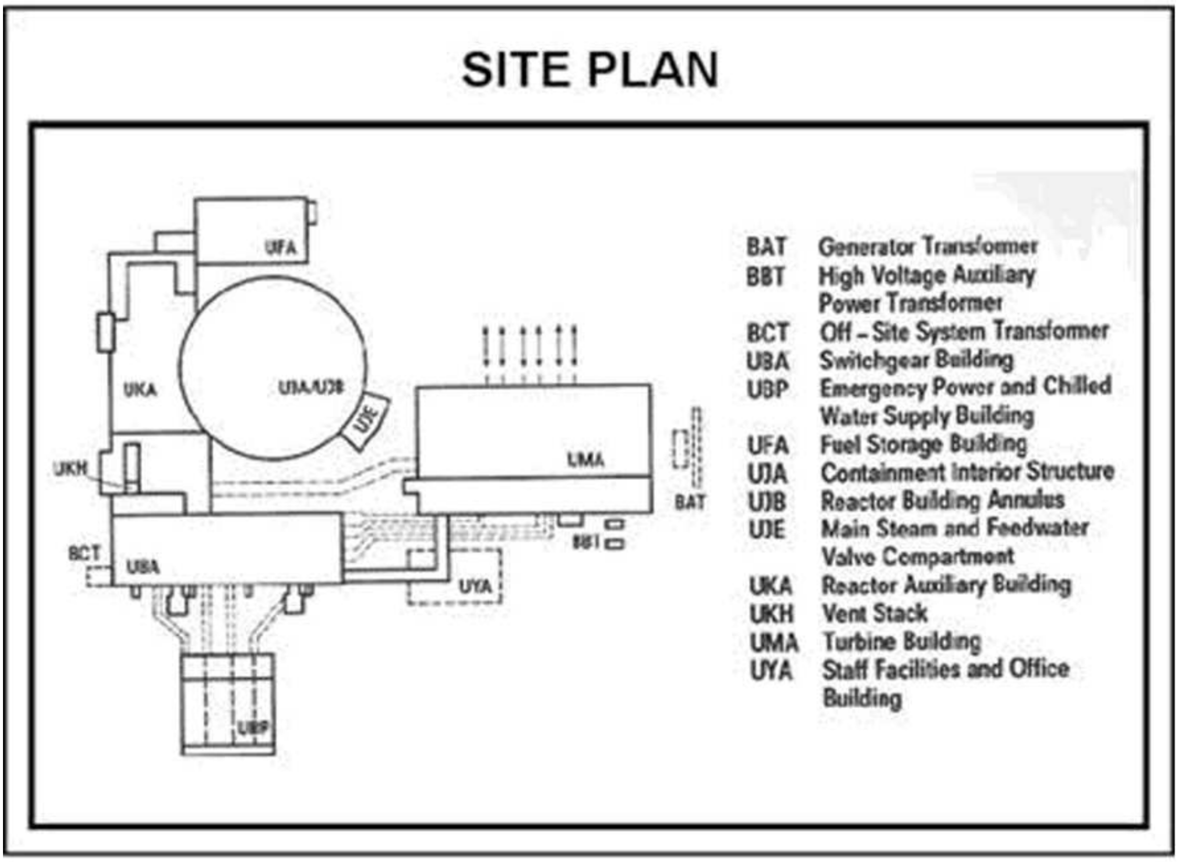


Figure III.3-3 - Atucha II Nuclear Power Plant - Main Building and Structures

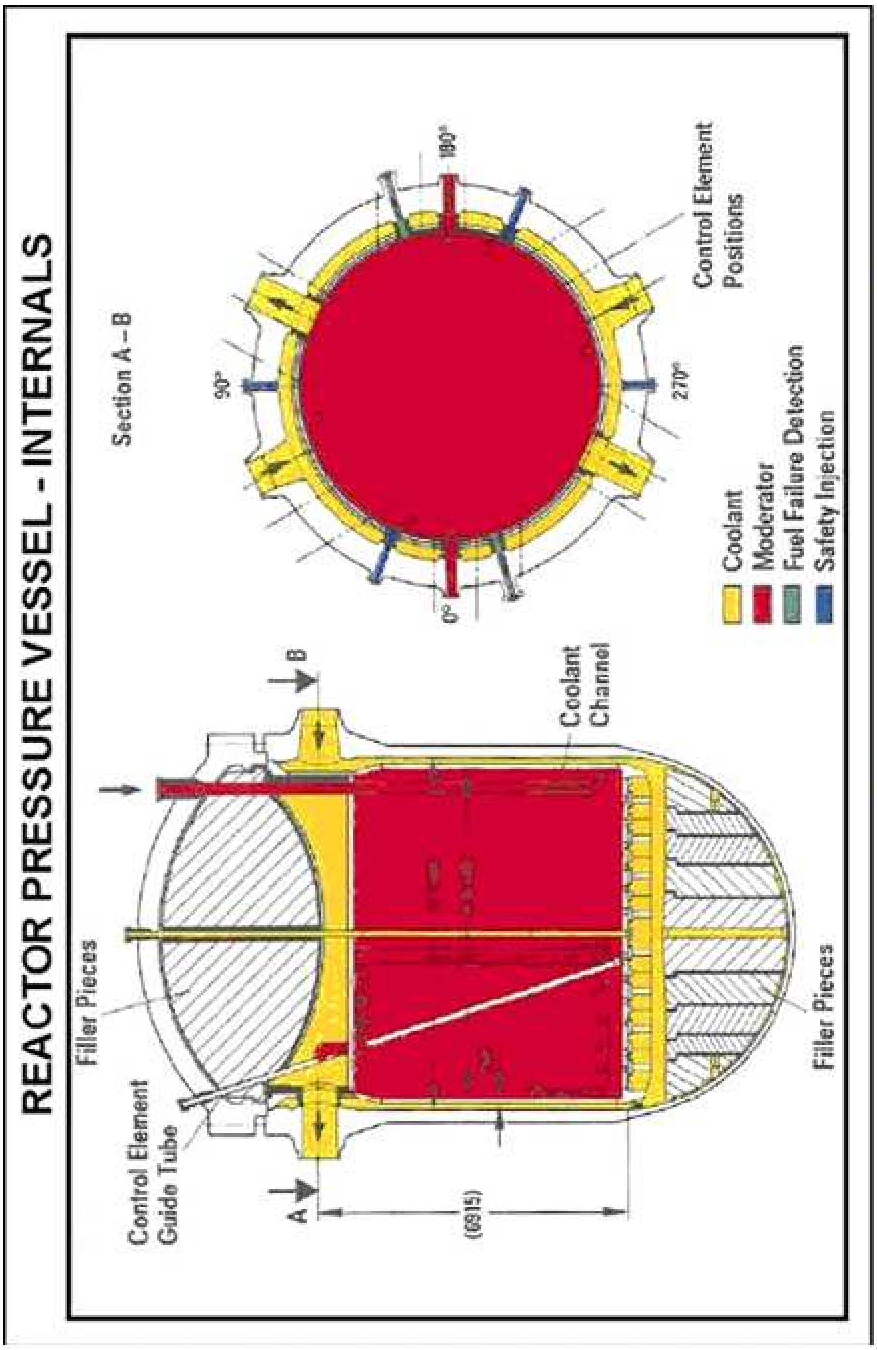


Figure III.3-4 - Atucha II Nuclear Power Plant - Reactor Pressure Vessel with Internals

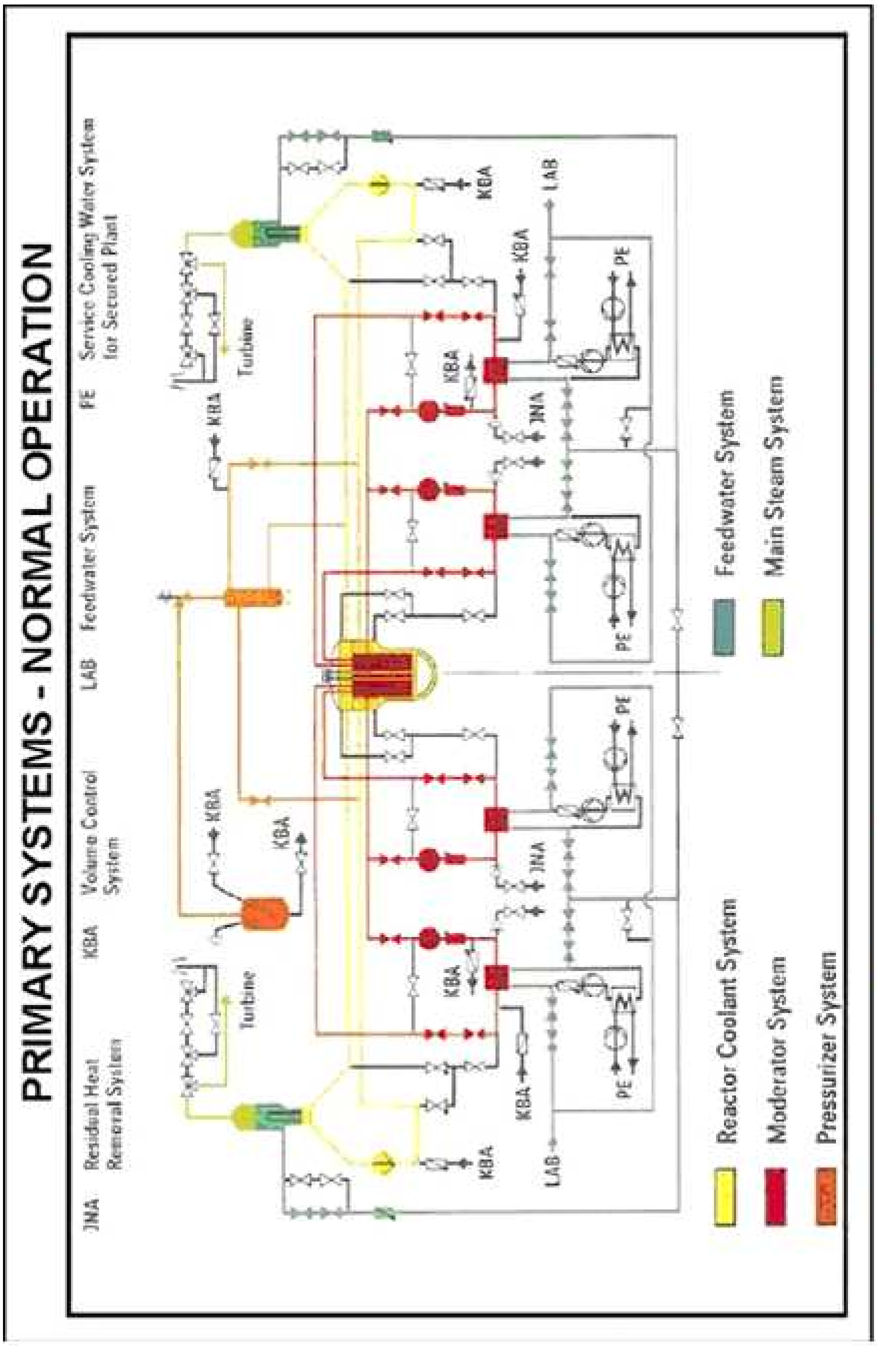


Figure III.3-5 - Atucha II Nuclear Power Plant - Primary Systems - Normal Operation

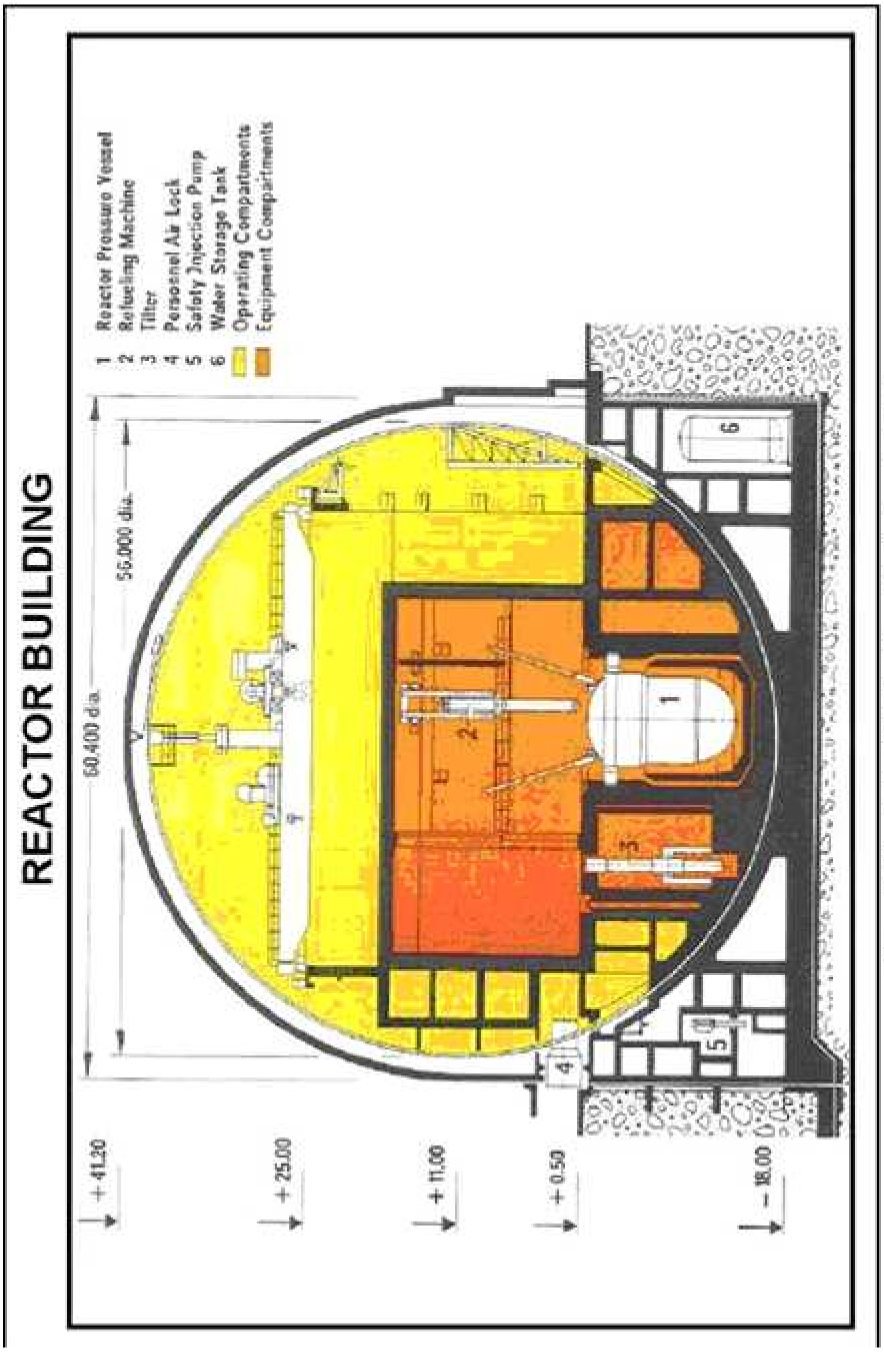


Figure III.3-6 - Atucha II Nuclear Power Plant - Reactor Building

ANNEX IV

PRINCIPAL TECHNICAL CHARACTERISTICS OF CAREM PROTOTYPE REACTOR

IV.1. INTRODUCTION

CAREM prototype reactor (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 ℃ 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 Residual 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.

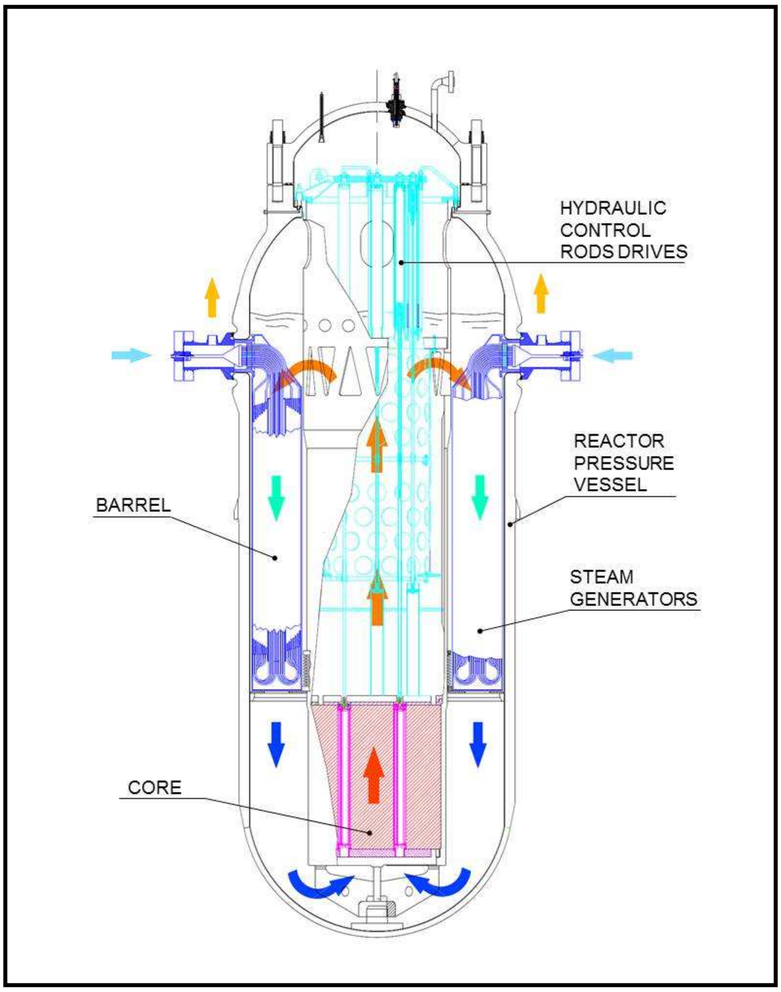


Figure IV.1 - Reactor Pressure Vessel

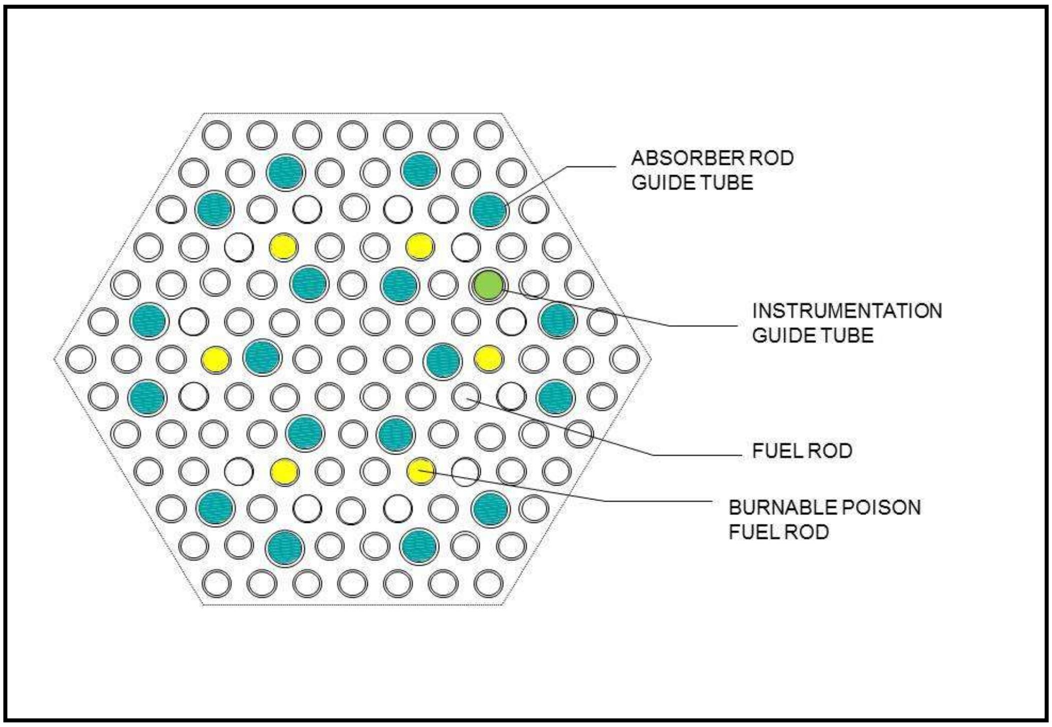


Figure IV.2 - Fuel Assembly Diagram. Fuel rods, guide thimbles and instrumentation thimble distribution

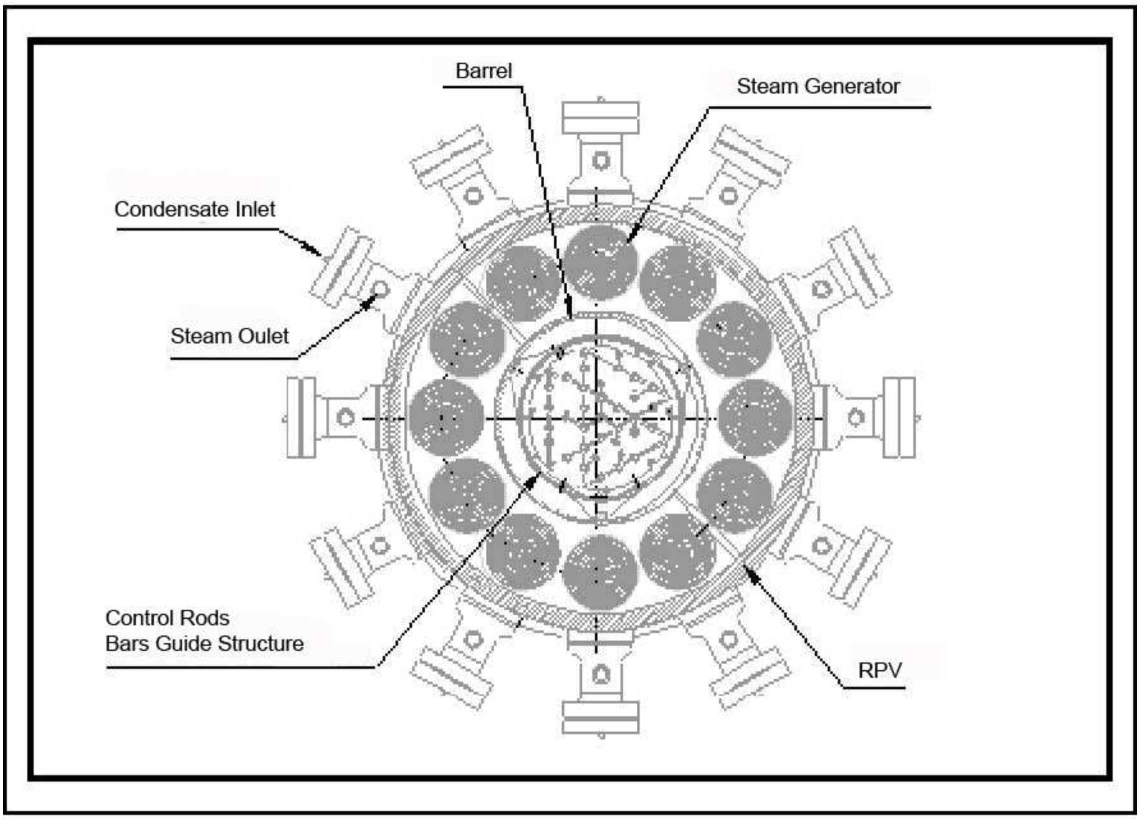


Figure IV.3 - Steam Generation Layout

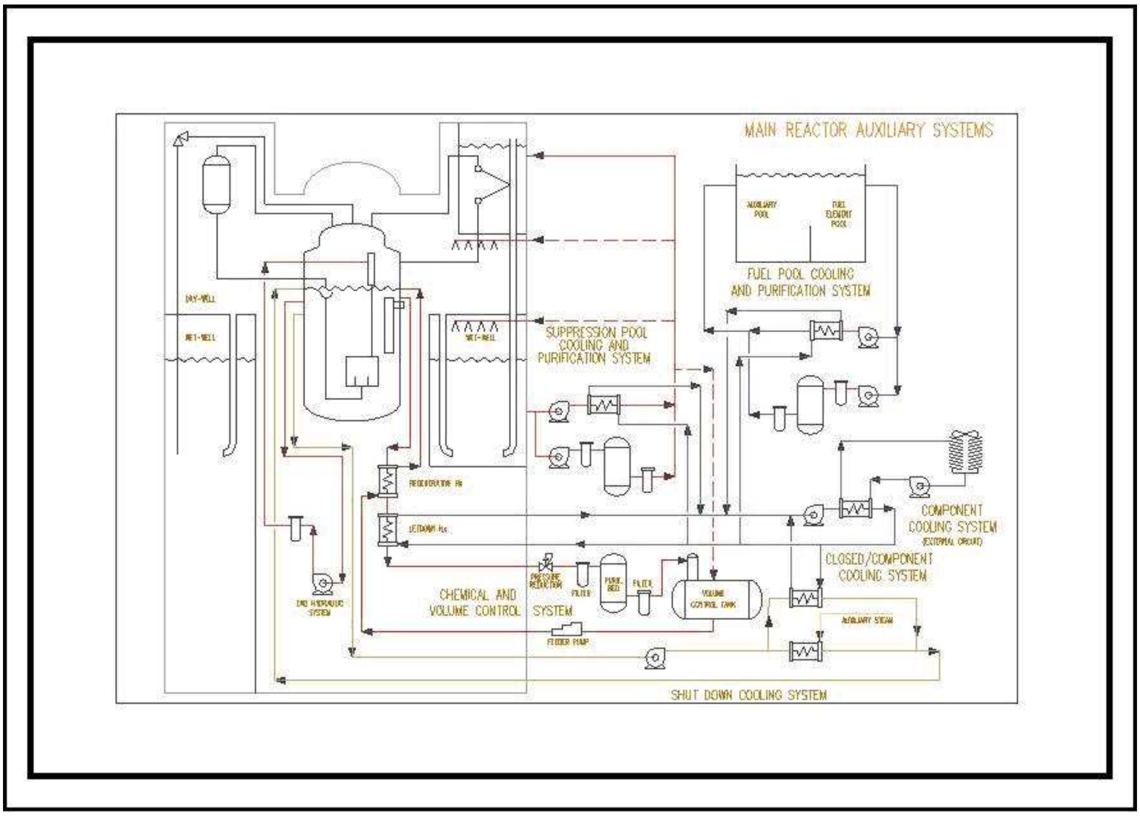


Figure IV.4 - Auxiliary Systems

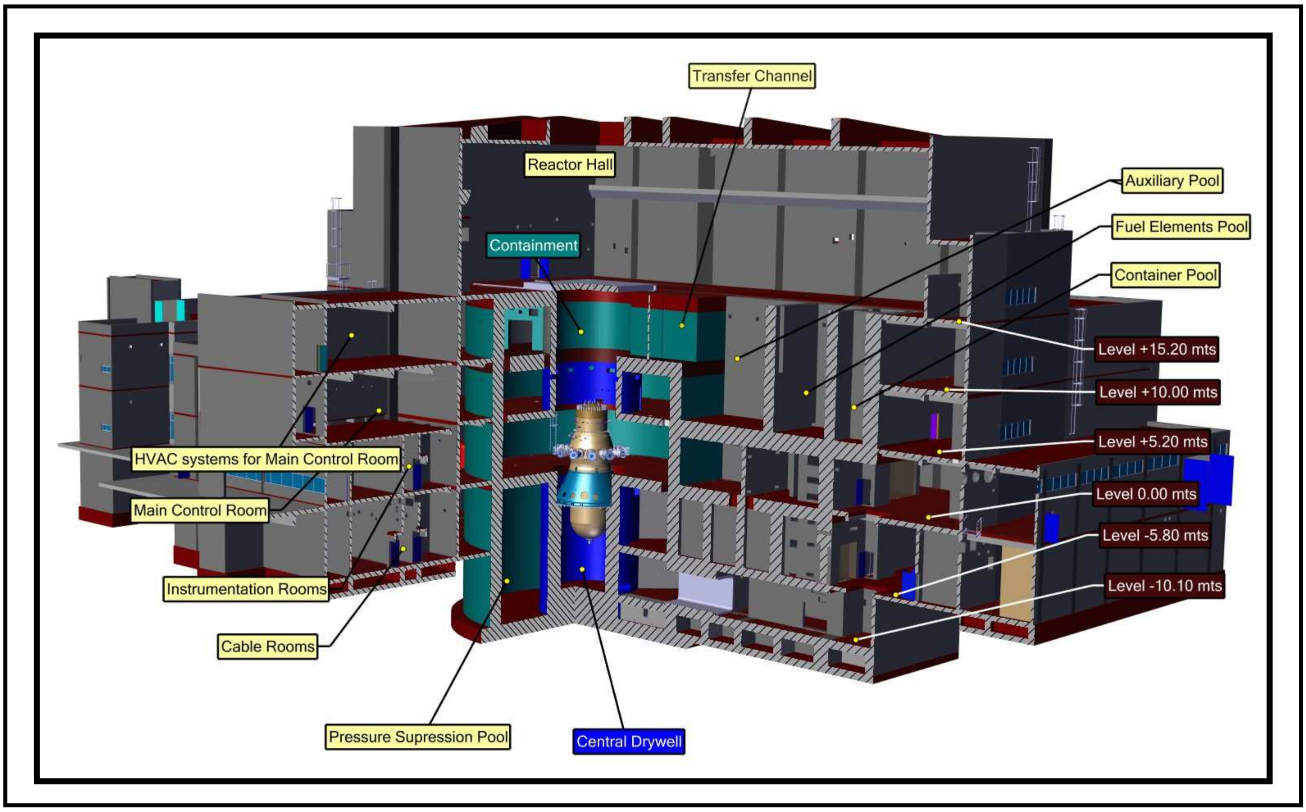


Figure IV.5 - Plant Layout

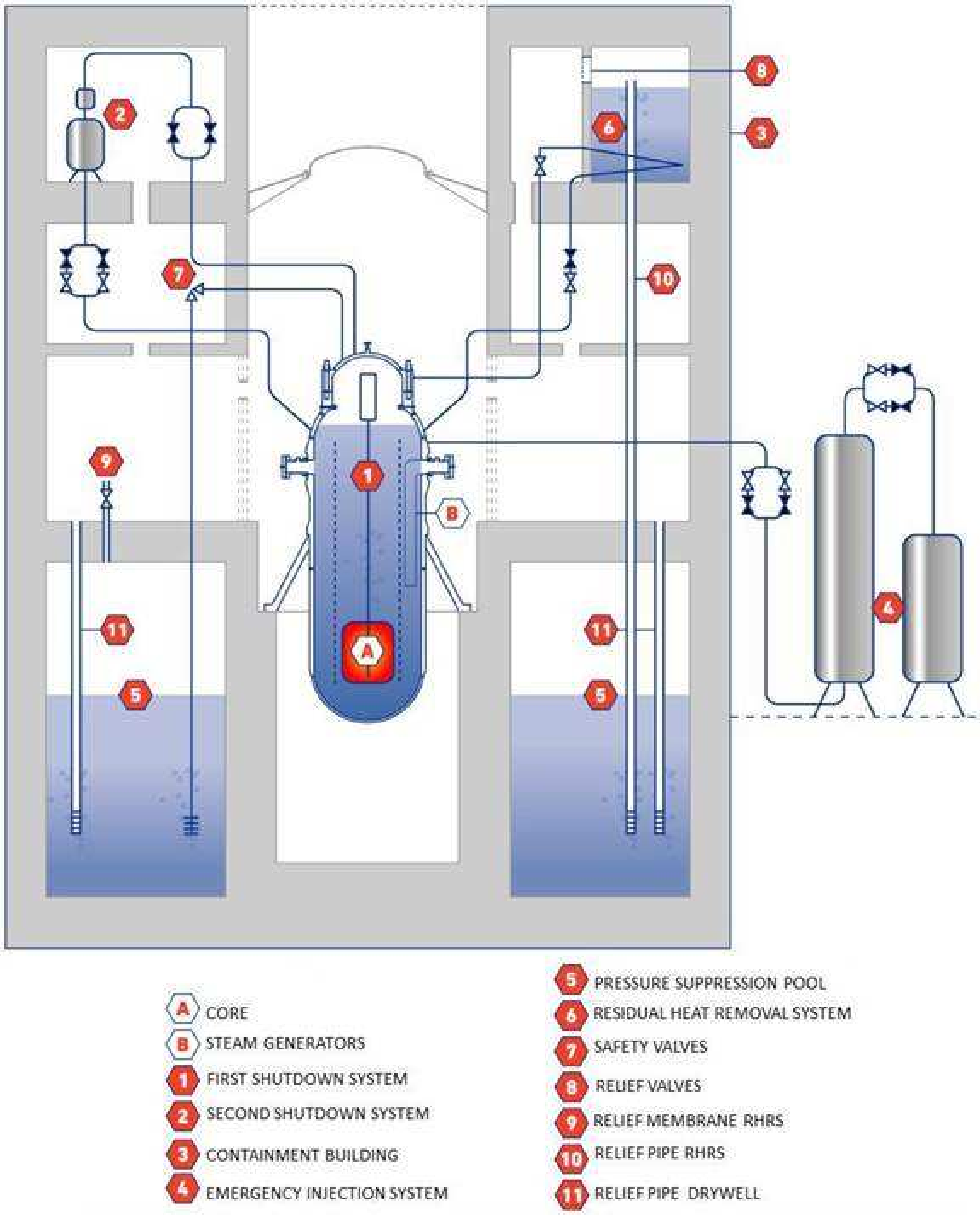


Figure IV.6 - Containment and Safety Systems

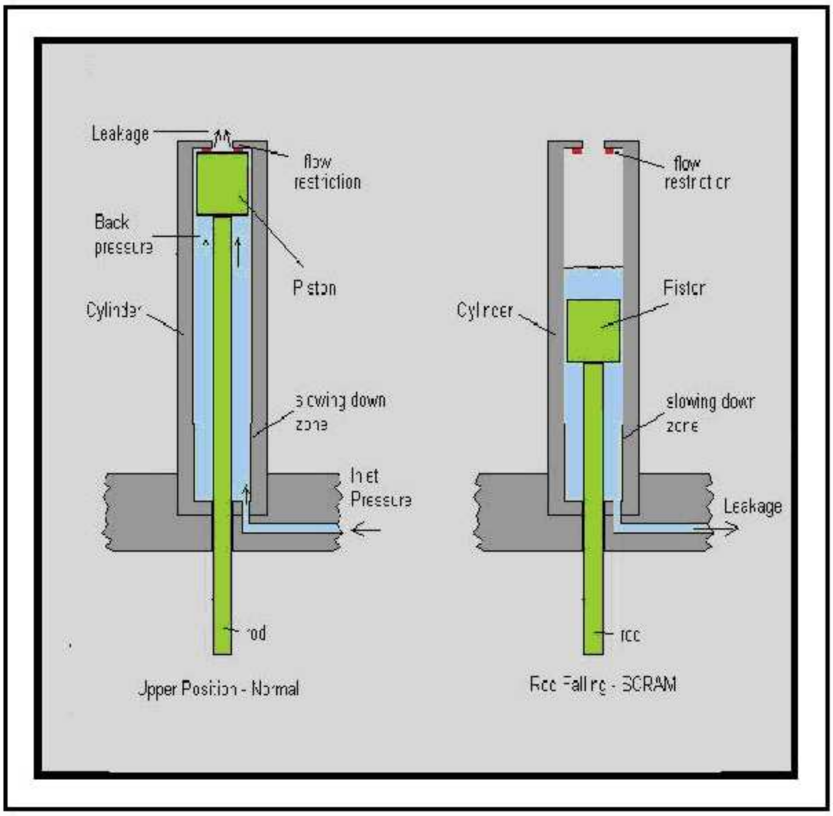


Figure IV.7 - Simplified Operation Diagram of a Hydraulic Control Rod Drive (Fast Shutdown System)

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 2019 – March 2022*, are listed, and how the licensee 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
DOUBLE FENCE BREAKAGE IN THE RADIOACTIVE WASTE DEPOSIT (EMBALSE - PHWR - 2022/03/31)	*CNA I-II: Disseminate the event emphasizing that the design authority is Plant Engineering.*
FATAL INJURY OF CONTRACTOR WORKER FOLLOWING THE FALL FROM THE ROOF OF AN AUXILIARY WAREHOUSE IN THE TURBINE HALL (DUKOVANY 1 - PWR - 2022/02/03)	*CNA I-II: Disseminate the lesson learned in interested areas.*
SOURCE RANGE HIGH FLUX REACTOR TRIP BYPASSED WHEN REQUIRED (BRAIDWOOD 1 - PWR - 2021/12/16)	*CNA I-II: Disseminate the event to Operation staff of both units.*
AN AUTOMATIC REACTOR SCRAM DUE TO THE ABNORMAL CLOSURE OF A MAIN STEAM ISOLATION VALVE CONTROL SWITCH CAUSED BY A MOVING CHAIR (KUOSHENG 2 - BWR - 2021/12/16)	*CNA I-II: Inclusion in Operational Annual Retraining.*
A CONTRACTED MAINTENANCE WORKER FATALLY INJURED (KURSK 3 - LWGR - 2021/11/09)	*CNA I-II: Disseminate the event through the Plant TVs.*
REACTOR SCRAM AND SAFETY INJECTION DURING COMMISSIONING (Common event - 2021/09/21)	*CNA I-II: Include the event in the Operation training of both units.*
EXCAVATING AND DRILLING WORKS (Common event - 2021/08/06)	*CNA I-II: Disseminate the event to the manager board and to the head and the second of the General Maintenance Services Department.*

OPERATOR-INDUCED EVENTS (Common event - 2020/12/28)	*CNA I-II: Disseminate the event emphasizing that it would be beneficial for supervisors and operators, when assigning tasks to employ effective mitigation strategies to ensure the success of the task; and that when preparing for a task, operations supervisors should ensure that the level of guidance contained in the operating procedures is commensurate with the skill level of the assigned operator.*
WATER HAMMER EVENTS (Common event - 2020/09/28)	*CNA I-II: Include in the Operation training of both units, the "water hammer" issue. Reinforce this event to the interested areas.*
USE OF UNAPPROVED DIFFERENTIAL PRESSURE TRANSMITTERS (Goesgen 1 - PWR - 2019/12/03)	*CNA I-II: Train the areas that intervene equipment, about the equipment qualification concept (seismic, environmental, radiofrequency). Define scope.*
PREVENTING DEBRIS-INDUCED FUEL FAILURES (Common event - 2019/08/12)	*CNA I-II: Disseminate the event in Nuclear Safety department.*
FOREIGN MATERIAL RETRIEVED FROM FUELLING MACHINE B-RAM FILL LINE FLOW ORIFICE (Point Lepreau, PHWR, 2019/11/19)	*CNE: Inclusion in Annual Retraining Programme for staff with specific authorization.*
HIGH PRESSURE COOLANT INJECTION AND REACTOR CORE ISOLATION COOLING INOPERABLE WHEN REQUIRED BY TECHNICAL SPECIFICATIONS (Hatch 2, BWR, 2020/04/22)	*CNE: Inclusion in Annual Retraining Programme for staff with specific authorization.*
OPERATOR-INDUCED EVENTS (Common event - 2020/06/02)	*CNE: Inclusion in Annual Retraining Programme for staff with specific authorization.* *CNA I-II: Disseminate the Event within Operations Staff emphasizing that when tasks are assigned, supervisors and operators should use effective mitigation strategies.*
LEAK OF ACTIVE WATER TREATMENT PLANT RESIN INTO FLOOR DRAINS RESULTING IN 4 MAN MSV UNPLANNED DOSE DURING CLEAN UP (Olkiluoto 2, BWR, 2020/03/03)	*CNE: Inclusion in Annual Retraining Programme for staff with specific authorization.*
WANO PARIS CENTRE ENGINEERING FLEET ANALYSIS (Common event - 2020/12/01)	*CNE: Procedure PG-1009 ("CNE definitive design change modification implementation") modification in order to add that Operations and Maintenance Managers approve basic engineering type B and C design changes in the frame of System Health Committee.*
FATAL INJURY OPERATOR THROUGH UNEXPECTED DISCHARGE OF INACTIVE AUXILIARY STEAM (Grohnde 1, PWR, 2016/08/26)	*CNE: Inclusion in Annual Retraining Programme for staff with specific authorization.*
USE OF CRANE NEAR 345-KV TRANSMISSION LINE CAUSED ARC AND TURBINE TRIP FOLLOWED BY REACTOR TRIP (Kori, PWR, 2021/04/23)	*CNE: Inclusion of event report in Work Order No. 155601 ("Request mechanical maintenance assistance with TEREX crane when lifting ET10, ET6 and ET1 HVAC motors to turbine building roof. Lifting plan is added").*

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. *Plant shutdown due to failure of an emergency shutdown safety system valve, during a surveillance test (CNA I, June 14th, 2019)*

On 2019/06/14, with unit I at 100% steady power during the surveillance test on the emergency shutdown safety system, a quick opening valve did not open when requested. The valve enables air pressurizing over the deuteroboric acid solution tank on loop 20 allowing boron solution injection inside the reactor core, in order to have a quick reactor shutdown. The valve failure led the plant to enter in a Limit Condition of Operation (LCO) and subsequently to a manual plant shutdown for 48 hours.

Direct Cause: the valve inner shaft came loose from its nut due to wear. The loose shaft was displaced until it was stuck on the valve opening seat, preventing it from opening.

Root Cause and Contributing factors: failure to verify the internal components condition in the previous maintenance because the maintenance instruction did not request this verification. The quality control during the valve reassembly in the previous maintenance was not performed properly.

Lessons learned / Corrective actions:

- *Update the maintenance valve reassembly instruction to request the checking of the inner shaft and nut condition before assembly.*
- *Retrain the Maintenance staff on the expectation of component condition check.*
- *Retrain the Quality Control staff on the importance of QC while tasks are being performed.*

V.2.2. *Complete loss of offsite power after Argentine Interconnection System breakdown (CNA, June 16th, 2019)*

On 2019/06/16 at 07:06 with unit I in hot shutdown and unit II at 40% power, a complete loss of offsite power (lines of 500 kV, 220 kV and 132 kV) occurred as a result of the national grid interconnection system (SADI) breakdown. Unit I remained fed by diesel emergency generators until 11:32 hours. Unit II underwent a load rejection and remained on house load operation until 11:00 hours, when it was reconnected to the 500 kV line. Both units responded according to design, without significant anomalies and the human performance was satisfactory.

Root Cause: The causes of the SADI breakdown were external to the Site and beyond plant management control.

Lessons learned / Corrective actions:

Besides immediate actions established in emergency procedures, maintenance on duty was established in order to attend to the Diesel Generators and some actions were taken to ensure fuel availability for, at least, 12 hours of Diesel generators operation.

At the Site, minor anomalies or other unexpected situations occurred, and corrective actions were issued:

- *Implement diverse safety electrical feed for the 'Security and Access Control' System boards distributed among Units I and II.*
- *Implement diverse safety electrical feed for the Reception Building.*

Due to this event, the Emergency Response Organization (ORE), was preventively called for the first time for a real issue. There were a few topics for which improvement actions were implemented:

- *Manage diverse phone services though mobile phone companies in order to diverse and increase emergency call outlets.*
- *Design robust communication alternatives with ORE staff, in order to get diversity and redundancy in the call method to prevent its exclusive dependence on private mobile phone services.*

V.2.3. *Controlled manual shutdown following elevate tritium caused by a crack in a moderator pump housing (CNA II, June 25th, 2020)*

On 2020/06/25 a controlled unit shutdown of unit II was initiated from 70% full power in order to search for and repair a leak following indications of elevated tritium concentration in the reactor and steam generator building. This leak was being monitored and, as it was increasing, it was decided (by ODM process) to preventively shutdown. A small crack measuring about 1mm was found at the upper weld ring of the junction of the cooling housing with the pump upper body on a moderator cooling pump.

Direct Cause: Crack in the moderator's pump housing weld caused by thermal shock as a result of an intervention in a nearby component that caused high stresses in the weld of the moderator pump housing.

Root Cause: An unrecorded repair performed during the commissioning stage near to the fissure, which was not identified during initial installation, left residual mechanical stresses in the surrounding area of the pump housing weld.

Lessons learned / Corrective actions:

- *In order to evaluate thermal stresses in the moderator pump housing during the cooling and heating processes, strain gauges and thermocouples were installed.*
- *An operation document (service order) was issued to enable the KAA system to cool the moderator pumps, in order to minimize thermal stresses on the pump bodies.*
- *Perform new penetrant ink tests to the weld rings of all moderator pump housings at the next scheduled outage.*
- *Perform a fractomechanical analysis with various fissure types to determine if and at what time (thermal cycles), the minimum detectable initial fissure can leak.*
- *Analyse the results of the strain gauges and thermocouples installed and, consequently, correct/validate the stress analysis performed.*
- *Include the service order issued in the Operation Manual.*

V.2.4. *Post start-up manual plant shutdown due to tritium leak in a safety valve, after maintenance (CNA I, August 12th, 2020)*

On 2020/08/12 with unit I increasing power (70%) after a scheduled maintenance outage, an increase in tritium concentration was detected in the steam generators room. Since the entrance to this room is prohibited when the plant is in service, it was decided to manually shutdown it for inspection. Then, a steam fume was visualized coming from a safety valve that connects the pressurizer of the primary cooling system with the pressure relief tank of the pressurizer. This valve had been previously intervened as preventive maintenance, but during its final tightening, one bolt remained less tight than the others. Also a foreign material must have pierced the rupture disc, which is placed on the valve seat.

Root Cause and Contributing factors: Lack of proper checking of tightening, due to less experienced work team, and a narrow scope of the task instruction. Although the task instruction requests to clean the valve seat surface before assembly, this step is not detailed enough to ensure preventing any foreign material from piercing the rupture disk.

Lessons learned / Corrective actions:

- *Modify the task internal instruction by adding the following steps: 1) Include an additional measurement of the flange gaps when tightening is being carried out, in order to know if the flanges are being tightened properly. 2) Include a table where all measurements must to be recorded to allow corresponding comparisons. 3) Add more detailed steps on cleaning the valve seat surface before assembly. 4) Add a maximum allowable value of flange gaps to be checked during tightening.*
- *Purchase a specific torque wrench that is best suited for this specific task.*
- *Train the mechanical maintenance staff in the workshop, emphasizing proper work practices for this task (as well as for the same valves on the other loops), and the need to double check the critical steps.*

- *Disseminate the event among the mechanical maintenance staff, emphasizing in the task supervisory and the need to highlight and clarify the critical steps in the pre-job briefings.*
- *Disseminate the event among the Quality control staff, emphasizing the need for awareness and control of the critical steps of the task.*

V.2.5. *Controlled manual shutdown for maintenance due to foreign material entry (CNA II, January 03rd, 2021)*

While Unit II was at 40% and increasing power after an outage, a higher than expected hydraulic resistance was detected in the primary cooling system. Therefore, the unit was shutdown to investigate and foreign material (FM) stuck in the internals of a recently maintained main cooling pump was observed. After analysis, it was concluded that the FM was an adhesive plastic tape material (TESA), which was used during maintenance, which had fallen into the open body of the pump during maintenance. As a result, the plant remained in shutdown state for 56 days, and after restart, the power increase ratio was lower than usual due to a need of monitoring the debris effect.

Direct Cause: FM intrusion into the opened system during an outage.

Root Cause and Contributing factors: Defective implementation of the Foreign Material Exclusion (FME) process. Inadequate work control and check out.

Lessons learned / Corrective actions:

- *Generate a plant start-up strategy that takes into account the presence of a greater amount of debris in the primary system.*
- *Retrain all plant staff on the consequences of incorrect implementation of FME program.*
- *The instructions for disassembly / reassembly of the main cooling pumps were analysed, and the weaker steps regarding FME were identified in order to be improved.*
- *Modification of the instructions for disassembling / reassembling the main cooling pumps, in order to strengthen the weak steps of the FME program, including among other items: the request for a final FME pipes inspection and the implementation of video cameras to monitor and record the system open spots all the time during maintenance.*

V.2.6. *Automatic SCRAM due to grid-related circuit breaker failure to open after the loss of the 220 kV external power line (CNA I, April 04th, 2021)*

With unit I operating at 100%, a failure in the external electrical grid led to a block protection occurrence and a turbine trip, followed by the reactor automatic SCRAM, and the activation of the Emergency Power Supply System, even without the existence of other failures in equipment or systems (See V.2.7). The cause of the event was the failure to open the circuit breaker at the 220 kV external power line end, because it did not receive the tele-protection signal after the line failure (due to contamination of insulators with bird droppings). As a consequence, there was no separation between the electrical block and the faulty external line, thus the overcurrent block protection was triggered, producing the turbine trip by design. Then, the steam pressure increased rapidly, which produced a level drop in the steam generators, reaching the reactor SCRAM level. The electrical loads automatically transferred across in slow mode resulting in separation of the emergency busbars from the normal bus bars and tripping two reactor coolant pumps. The plant was stabilized utilizing the 132 kV external power supplies while the 220 kV external power line was restored.

Root Cause and Contributing factors: the cause of the event was an external grid failure mode that led directly to a plant shutdown. The behavior of the installation was according to design. "Slow" switching to 132 kV external power line: the fast switching device, having no reference voltage or phase on the 220 kV side (due to line failure), switched in "slow" mode by design. No events have been recorded previously in which the switching maneuver has been made under similar conditions.

Lessons learned / Corrective actions:

- *A preventive maintenance plan was requested to the grid operator for circuit breakers AC01, AC02 and their teleprotections, as they proved to be vulnerable points for the availability of the plant.*

- *Request the corporate's safety and design group to evaluate the SCRAM trip criteria for low level in the steam generators in order to prevent pressure transients from reaching the reactor SCRAM value due to bubble collapse in steam generators (Improvement action).*
- *Carry out an analysis of the block failure switching criteria to avoid triggering the Emergency Power Supply system (Improvement action).*

V.2.7. *Emergency Power Supply system actuation after power block protection trip and external power line switching pipe coupling (CNA I, April 04th and 06th, 2021)*

During two events occurred on 2021/04/04 and 2021/04/06,the reactor automatic SCRAM was produced after the trip of the electrical block protection which caused the switching of the external electrical power supply from the 220 kV line to the 132 kV line. During the switching transient, the activation of the Emergency Power Supply System (EPS) on both trains was unexpectedly produced despite the availability of offsite power. In both events, the fast switching device, switched in "slow" mode (the "fast" mode, which must be completed in less than 100 ms, avoids not only the automatic SCRAM, but also the activation of the EPS).

Root Cause: In 2015 the old EPS was replaced by a new one, introducing a relevant change in the condition of the emergency busbars BU and BV. Previously they were considered as uninterruptible, and now they are allowed to have a short interruption and operating decoupled in all operating states. When designing the new EPS, the 100 ms criterion for opening the coupling between normal and emergency busbars was not modified, although it was no longer necessary.

Lessons learned / Corrective actions:

- *Carry out, during plant shutdown, the revision of the BB bus switching from 220 kV to 132 kV.*
- *Perform engineering analysis and documented proposal to eliminate the opening of BA101 / BB101 at 100 ms.*

V.2.8. *Rework due to a no-watertight moderator pump vent pipe coupling (CNA I, October 16th, 2021)*

On 2021/10/16 after unit I reached the hot shutdown status from the cold shutdown, an increase in tritium concentration was observed in the steam generators room. A heavy water loss was found, coming from a no-watertight RT coupling from the continuous vent line of the moderator pump. The plant was taken to cold shutdown without pressure for the intervention. The consequences were a delay in returning to the hot shutdown state, additional personnel dose (61 mSv cumulative dose) caused by rework, and heavy water spill.

Direct Cause: The RT coupling was erroneously loosened and then not readjusted.

Root Cause and Contributing factors: The coupling was not included in the adjustment checklist and inadequate drawings because the RT coupling did not appear on any of the pump detailed drawings. When it was loosened, there was no questioning attitude of the personnel who carried out the tasks regarding the components that they really had to loosen.

Lessons learned / Corrective actions:

- *Modify the internal instruction for the moderator pump QP01D001 replacement in order to include the experience gained, adding the connection checklist.*
- *Adapt the QP01D001 pump and primary system drawings adding the missing RT couplings.*
- *Disseminate the event to the entire Maintenance area emphasizing having a more questioning attitude and reporting unexpected situations that differ from planned tasks. Also, reinforce the expectation of the role of the person responsible for the task.*

V.2.9. Manual turbine trip due to incorrect protection parameters set during refurbishment (CNE, March 09th, 2019)

During normal operation and while performing a turbine generator (TG) reactive power change, the TG protection was initiated by an over excitation current trip and resulting in a TG trip. The reactor power was shut down and the power reduced to 10%.

Direct Cause: Inadequate design and implementation of revised generator protection settings.

Root Cause: Inadequate design analysis of the generator protection scheme settings.

Lessons learned / Corrective actions:

- *Perform the required modifications that arise from the Faults analysis regarding the Generator's electrical protections.*
- *Inform the manufacturer about the deficiencies found in the "Protections' Adjustment and Configuration" study.*
- *Spread the event among the involved personnel.*

V.2.10. Automatic reactor scram due to high neutronic power caused by electrical noise (CNE, April 25th, 2019)

During normal operation following a refurbishment outage, the reactor protection initiated on high neutronic power signals resulting in an automatic scram and an outage of four days.

Direct Cause: Noise generation in the neutron power detectors due to inadequate grounding.

Root Cause: Inadequately conceived design changes to the recent reactor over power protection including the lack of verification.

Lessons learned / Corrective actions:

- *Include the following items in an existing Deficiency Report for "Shutdown System #2 enhancement": Study of the grounding connections of the SDS2, verifying the existing documentation; in case that "in-field verifications" are required, then create work orders to arrange survey tasks.*
- *Spread the event among the involved personnel in the SDS2 design change process, reinforcing the concept of Independent Verification in these processes.*

V.2.11. Trip of shutdown system due to steam generators low level during "instrument installation and enabling" performance test (CNE, May 14th, 2019)

During commissioning at full power following an extensive refurbishment outage and while installing instrumentation for a performance test, an erroneous signal from a turbine pressure transmitter caused inadvertent closure of the turbine steam control valves. An automatic reactor power reduction occurred, followed by a reactor scram on low steam generator level.

Direct Cause: Incorrect installation of the instrumentation for the test.

Root Cause: Inadequate risk assessment of the activity, which failed to identify that the pressure transmitter is a single point vulnerability component.

Lessons learned / Corrective actions:

- *Re-evaluate the plant instrument to be intervened during the Performance Test period (Consider signal inhibition during the intervention).*
- *Include the Pressure Transmitter "PT71" in the Master Equipment List as a "Single Point Vulnerability" equipment.*

V.2.12. Turbine trip due to high level of water in Steam Generator #3 (CNE, May 18th, 2019)

During normal operation and during a routine changeover of boiler feed pumps for filter maintenance, a steam generator level fell. The reactor power was reduced to 95% and during the operation a further level transient occurred initiating the turbine protection resulting in a turbine trip and a reactor setback to 45%.

Direct Cause: The cause of the first transient was due to a recirculating flow control valve failing to open due to a loose air supply hose. The cause of the second transient was due to a level control regulating valve failing to open due to inadequate lubrication.

Lessons learned / Corrective actions:

- *Verify the correct performance of the Level Control Valve of the Boiler #2 "LCV-2A1", and re-check the "LCV3A1" valve.*
- *Perform an inspection and modification of the system's Pump Recirculation Flow Control Valves #82 (Pump "A"), #86 (Pump "B"), and #90 (Pump "C") in order to prevent air hose loosening. Perform a routine survey until modification.*

V.2.13. Complete loss of offside power after national grid system breakdown (CNE, June 19th, 2019)

During commissioning following an extensive refurbishment outage and while at 99.5% power, a complete loss of offsite power was experienced initiating an automatic reactor power reduction to 60%. The turbine tripped resulting in the loss of power to the primary system pumps, the initiation of the reactor shutdown system. This event is noteworthy due to the total loss of offsite power for six hours.

Direct Cause: The loss of offsite power was caused by a fault on the national grid system. The cause of the turbine trip was a known fault on the turbine-generator digital electro-hydraulic control system which had been identified during testing resulting in the loss of house-load operation.

Lessons learned / Corrective actions:

- *Perform the necessary actions to ensure the proper functioning of the steam control valves in accordance with the implementation of the start-up procedure.*
- *Request the Grid Operator to guarantee the availability to connect to the Hydraulic Power Station "Reolin" in accordance to its internal procedure.*

V.2.14. Trip on low level in steam generator due to operator error (CNE, July 05th, 2019)

During normal operation, a steam generator level control valve was being calibrated while in manual control. The valve was returned to automatic control without balancing and reducing the control deviation, resulting in a rapid valve closure, the steam generator level drop, actuation of stepback and scram signals. The unit entered the planned outage a half day earlier than scheduled.

Direct Cause: Operator when switching the valve control from manual to automatic mode due to the inadequate preparation and control of the evolution.

Lessons learned / Corrective actions:

- *The dissemination of the event and the reinforcement of expectations regarding the preparation of the task and its potential consequences, as they are considered in the procedure regarding the levels of risks.*
- *The Engineering and Operations team were involved in training.*

V.2.15. Spent Fuel bundles deformation during discharge process into fuel pool (CNE, July 12th, 2019)

During normal operation and while performing on load refuelling, following the discharge of a fuel channel a fuel in air alarm was received requiring the entry into the emergency sequence procedure. All eight fuel bundles were transferred to the spent fuel pool. It was discovered that four of them were deformed due to inadequate cooling and drop of two fuel bundles to the pool bottom. The bundles were segregated and quarantined pending approval for further handling and storage. Further reactor refuelling was suspended. The event is categorised as significant due to the loss of cooling and drop of spent fuel assemblies during their transfer to the spent fuel pool, which resulted in their damage and created a potential for a significant fuel defect.

Direct Cause: The initial cause was a failure of a mechanical actuator to move the fuel far enough to reach the receiving ladle in the fuel pool reception bay and resulting in the receipt of the alarm. The apparent cause was the operator proceeded with the defuelling without confirming the cause of the alarm and failed to adhere to the emergency procedure.

Lessons learned / Corrective actions:

- *Open a deficiency report to include in the F/M Operating Manual what was expressed in the Operating Memorandum.*
- *Perform F/M operators training in the operation of Emergency Cooling Sequences.*
- *Disseminate the event in F/M Operators and Shift Supervisors (Immediate action).*
- *Disseminate the event in Fuel Management personnel.*
- *Normalize lighting of R-001 (Immediate action).*
- *Modify position, or add a video camera to see the download rack in R-001 enclosure.*
- *Evaluate the feasibility of modifying the Fuel Handling program to enter the verification of ball valve position of the spent fuel port, before starting each fuel bundle discharge sequence.*
- *Fix Ram 2 Force 2 failure (immediate action).*

V.2.16. Turbine shutdown to repair oil leak in the steam flow control valve from the turbine speed control system (CNE, July 12th, 2019)

During normal operation unexpected movement of a turbine control valve was observed. An oil leak from an associated servo valve was identified and monitored and the turbine was shutdown and reactor power reduced on indication of the leak rate increasing.

Direct Cause: The cause of the oil leak and the cause of the unexpected valve behaviour was not identified. The valve supplier at the next outage will perform additional fault analysis.

Lessons learned / Corrective actions:

- *Issue a Work Order for the next Planned Outage to inspect/replace the valve's Internals.*
- *Request a fault analysis to the valve supplier.*

V.2.17. Steam leakage from a feedwater system vent pipe (CNE, February 08th, 2020)

During operation at 80% a steam leak was discovered on a feed water system header and the reactor was shut down for repairs incurring an outage of just over two days.

Direct Cause: Weld failure at the weld connecting a feed system vent pipe and sockolet.

Root Cause: Failure to install pipework supports following a pipework design modification change at the previous outage.

Lessons learned / Corrective actions:

- *Send the event report to the designer. Request a response that includes failure analysis and a new design verification (for the fifth preheater train) to identify those smaller diameter pipes joining the collector that do not have supports.*
- *With the designer's response report (CA1), proceed to open the necessary work orders to place supports in those pipes with a smaller diameter than the header that have been identified, as necessary.*
- *Execute the work orders generated from CA2.*
- *Spread the event among Engineering, Operations and Mechanical Maintenance personnel.*

V.2.18. Unplanned outage due to low level in steam generator due to a flaw in the electro positioner (CNE, July 26th, 2020)

During normal operation the steam generator (SG) level on one SG started to drop. The load was initially reduced to 80%, the SG level continued to reduce. The load was further reduced and eventually the turbine was tripped at 5% power and the reactor was taken to hot shutdown.

Direct Cause: Failure of the feedwater flow control valve to match the required feed flow due to the presence of dust within the housing of the electro-positioner.

Lessons learned / Corrective actions:

- *Issue a Deficiency Report to analyze the operation of the electropositioners and their eventual replacement.*
- *Generate a work order to Instrumentation and Control (I&C) Maintenance, to check the interior of the housing of the electropositioners of the level control valves.*
- *Disseminate this event among I&C Engineering, I&C Maintenance and Operations personnel.*

V.2.19. Unplanned power reduction due to an obstruction in the hydraulic system for a turbine control valve actuator (CNE, December 21st, 2020)

During normal operations, following a turbine valves test, one low pressure turbine control valve unexpectedly closed due to a hydraulic actuation system failure. The power was automatically reduced to 45%.

Direct Cause: Flow restriction in the hydraulic system due to debris and impurities.

Root Cause: Inadequately performed preventive maintenance by installing an orifice having inappropriate dimension.

Lessons learned / Corrective actions:

- *Issue work orders to verify the flow restrictors in the other Low-Pressure Turbine's Flow Control Valves during the Planned Outage of 2021.*
- *Issue work orders to inspect and clean the servomotors.*

V.2.20. Power reduction caused by power load-unbalance signal due to failure in a Low-Pressure Turbine valve (CNE, January 24th, 2021)

During normal operation, after switching the electro hydraulic control system pump, one low-pressure turbine regulating valve and its related stop valve closed unexpectedly, resulting in an opening of the main steam safety valve and a reactor power reduction to 43% power.

Direct Cause: Inappropriate orifice diameter of a flow restrictor in the valve actuator's servomotor.

Lessons learned / Corrective actions:

- *To execute, during the Planned Outage of 2021, the work orders issued to inspect and clean the servomotors and to verify the flow restrictors in the other Low-Pressure Turbine's Flow Control Valves.*

V.2.21. Automatic SCRAM due to a spurious signal in digital electro-hydraulic control system (CNE, February 06th, 2021)

During normal operation, a spurious signal of very low pressure of digital electro-hydraulic control system occurred, resulting in a turbine trip. Following the plant transient, the reactor automatically scrammed on low boiler level.

Direct Cause: Water intrusion into a junction box of pressure switches used in the turbine trip logic.

Root Cause: Leak in an auxiliary valve of the steam generator feedwater system was not adequately managed.

Lessons learned / Corrective actions:

- *Develop a reinforcement program on Water Intrusion that includes the present event and the WANO SER 2016-2 training material.*
- *Adapt the orientation and sealing of the conduit that allowed the water ingress to the junction box.*
- *Create an inspection routine to check and clean the sumps/drains of the auxiliary building.*

V.2.22. Automatic SCRAM due to use of non-nuclear graded parts in a pressure controller (CNE, February 19th, 2021)

During operation at 100% power an unexpected power reduction to 17% within two minutes occurred without operator intervention. When the power reached 17% an automatic reactor scram occurred due to a steam generator low level.

Direct Cause: Loss of power to the helium balance header controller resulting in an increase in level within the liquid zone system.

Root Cause: Inadequate preventive maintenance programme. A contributing cause was inadequate management of temporary design changes within the controller power supply.

Lessons learned / Corrective actions:

- *Reinforce among the maintenance personnel on the importance of reporting the tasks carried out on the safety systems.*
- *Perform the complete replacement of the power supply in the controllers that had the task pending.*
- *Implement the analyzed modification to avoid unexpected trips of the Shutdown System #1.*
- *Spread the present event among Maintenance, Quality Control, Engineering and Work Package personnel. Carry out the reinforcement of: Questioning attitude regarding the instances of Independent Verification at the time of the closing of executed work orders: Each work order and temporary design change must be done by referencing the component.*

V.2.23. Controlled shutdown following normal operating procedure to carry out corrective maintenance due to break of feed piping of electrohydraulic control of a turbine control valve (CNE, March 06th, 2021)

During normal operation following a periodic test of the turbine emergency and intermediate valves an electro hydraulic oil leak was identified. The repair could not be performed on load and the reactor was shutdown.

Direct Cause: Failure of the intermediate pipe weld due to metal fatigue.

Root Cause: *Pipework vibration. A contributing cause was pipework support brackets that were outside of their designed position.*

Lessons learned / Corrective actions:

- *Verification and adjustment of the support of the EHC system. Execution of work orders during planned outage PP2021.*
- *Repair of the indications observed in the lines of some intermediate valves and some fluid control valves during the planned outage PP2021.*
- *To carry out a comprehensive study of the pipes, socket-welded unions and supports of the EHC system that supply oil to the turbine intermediate valves. To include the inspection of these components with a frequency to be determined as a compensatory action. To include evaluation, in the short term, of periodically including non-destructive tests on T-joint welds of the EHC system in the area of the intermediate valves and a periodic visual inspection of the supports.*
- *To transmit the event among personnel from Systems Engineering, Mechanical Maintenance and Operations. Reinforce the concept of T-junction metal fatigue as a consequence of system vibrations.*

V.2.24. Arcing in bus bar of auxiliary transformer causing damage, fire and injury due to operator error (CNE, April 07th, 2021)

During an outage, while installing a portable grounding equipment to tag-out an auxiliary transformer and 380 V bus prior to a preventive maintenance, an electric arc occurred, followed by flame causing hand burn injuries to one of the workers. The workers were transported to the plant medical facility for treatment. The emergency brigade extinguished the fire. The event resulted in an outage extension and eight day out of work of the injured worker. This event is categorised as noteworthy due to the strong potential for a more significant injury or fatality associated with electric shock.

Direct Cause: Human errors leading to identification of an incorrect transformer and incorrect use of electroscope.

Root Cause: Inadequate procedure adherence, inadequate verification and failure to use prescribed personal protective equipment. An additional cause was inadequate equipment labelling.

Lessons learned / Corrective actions:

- *Retraining in the Human Factors Simulator for the electricians involved in the event, on Error Prevention Tools, Industrial Safety (electrical risk), and good electrical practices.*
- *Reinforcement of work practices in electrical maintenance tasks for the shift electricians.*
- *Include the work of the shift specialist personnel (electricians, mechanics and instrumentalists) in the Task Observations program.*
- *Analyse a method for identifying electrical equipment in the Metal Clad, Power Centre and Auxiliary Transformers rooms that facilitate their identification.*

V.2.25. Automatic SCRAM due to spurious actuation of the liquid Injection shutdown system (CNE, September 28th, 2021)

During normal operation and while performing primary heat transfer (PHT) pressure guard line testing, two moderator high level trip signals were received initiating an automatic reactor scram.

Direct Cause: Spurious protection signals due to heat transfer between the PHT pressure and moderator high level trip sensor lines caused by a leaking PHT sensor line solenoid valve and exacerbated by the parallel routing and lack of thermal insulation on the sensor lines.

Root cause: Inadequately conceived design modification as both guard lines were installed at the recent life time refurbishment outage.

Lessons learned / Corrective actions:

- *Design change of the sample tubing, for separation/insulation of the measurement lines.*
- *By extension of the condition, perform an evaluation of the layout of the rest of the trip channels lines for Shutdown Systems #1 and #2.*
- *Modification of the Routine Test in order to reduce the time of execution, when executed within the frame of operational conditions that require the extension of the test, and modify the Operation's Field Procedures associated with the test.*
- *Replacement of the valve with an internal leak.*
- *Communicate the event among Operations, Engineering and Instrumentation and Control Maintenance.*

V.2.26. Perimeter fence damage of Radioactive Waste Deposit #7 (CNE, January 16th, 2022)

On January 16th, 2022, Unit I operating at full reactor power, a runaway incident of a gantry crane occurred during strong winds, damaging the double perimeter fence of the Radioactive Waste Deposit #7 with its associated volumetric-motion sensors and a video surveillance camera, leading to 16 hours of degraded monitoring of the area until detection. Contingencies were taken to monitor the Radioactive Waste Deposits perimeter while the gantry crane was removed and the fences, along with its volumetric-motion sensors and video surveillance, were restored.

Direct Cause: This event revealed weaknesses in the design and assembly of the gantry crane and Security operator fundamentals.

Lessons learned / Corrective actions:

- *Spread the event among Engineering, Production and Maintenance personnel, emphasizing the importance of rigorous control of the conditions established in the design during the implementation of modifications in the plant.*
- *Spread the event to Security personnel, and develop a training program for Security system operators, based on the reinforcement of the operator fundamentals.*
- *Maintenance and Industrial Safety will evaluate that all site's gantry cranes, verifying the existence and operating conditions of braking systems and, if necessary, fastening components.*
- *Include the verification of the correct operating status of outdoor gantry cranes' braking/fastening components in the procedure for severe weather alerts' walkdowns.*

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 NA-SA organizational chart.

Annex 3 Definitions.

Annex 4 Manual distribution lists.

Annex 5 Quality policy.

Annex 6 SGC documents and records.

VI.2. QUALITY AND ENVIRONMENT POLICY

Nucleoeléctrica Argentina S.A., 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 control of activities***

  Make a continuous effort to plan and control all Company activities that are directly or indirectly related to the safety and availability of its facilities.

- ***Protect the environment***

  Make a continuous effort to prevent pollution and minimize the adverse environmental impact derived from our activities. Operate the facilities making rational use of energy and natural resources.

- ***Adapt management to applicable laws and regulations***

  To comply with the legislation and regulations applicable to the different facilities and activities of Nucleoeléctrica Argentina S.A. and the other requirements subscribed by the organization.

- ***Promote business management by risks and opportunities***

  Evaluate the potential risks and opportunities of power generation activities and new projects.

- ***Promote staff training, coaching and knowledge management***

  Train and coach staff to ensure that they are competent to perform their 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 the environment. To take the necessary actions to safeguard the organization's knowledge.

- ***Encourage internal and external communication***

  Communicate this Policy to all staff ensuring its understanding and compliance. Publish it so that it is available to relevant stakeholders. Inform about the benefits of the nuclear option and its contribution to the preservation of the environment.

- ***Continuously improve quality and environmental management***

  To strive for continuous improvement and effective management through systematic and periodic evaluation of quality and environmental management, implementation of identified improvement opportunities and international nuclear industry excellence practices.

- ***Consider in its activities the needs and expectations of interested parties***

  Identify and understand the needs and expectations of stakeholders, developing the necessary actions to meet the requirements determined by the organization.

CONVENTION ON NUCLEAR SAFETY

Autoridad Regulatoria Nuclear
PRESIDENCIA DE LA NACIÓN