

NATIONAL NUCLEAR SAFETY REPORT **2013**



Autoridad Regulatoria
Nuclear

Presidencia de la Nación Argentina

ARGENTINEAN NATIONAL REPORT FOR THE CONVENTION ON NUCLEAR SAFETY August **2013**



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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)

CONTENTS

GLOSSARY	VII
INTRODUCTION	1
1.1. General concepts	1
1.2. National policy in the nuclear field	2
1.3. National program corresponding to nuclear installations	2
1.4. Summary of the main subjects contained in the Report	2
1.4.1. Actions taken in the light of the Fukushima Daiichi Accident	4
1.4.1.1. External events	5
1.4.1.2. Design issues / Analysis of loss of safety functions	6
1.4.1.3. Severe accident management and recovery –on site–	7
1.4.1.4. Emergency preparedness and response and post-accident management –off site–	7
1.4.2. Annexes	8
CHAPTER 2 – FOLLOW-UP FROM THE FIFTH REVIEW MEETING	9
2.1. The spent fuel storage at Atucha I NPP	9
2.2. Adequacy of training for staff for the Regulatory Body and knowledge retention as well as for the Licensee	10
2.3. Licensee human performance programs	11
2.4. NPPs' refurbishment and construction programs	11
2.5. Install full scope training simulators	12
2.6. External hazards assessments including flooding and seismic re-evaluation	12
2.7. Severe accident management	13
2.8. Use of international reviews	15
2.9. Plant safety upgrades	16
2.10. Strategic communication plan	20
2.11. Enhance the current strong relationships with other organizations	22
CHAPTER 3 – COMPLIANCE WITH ARTICLES OF THE CONVENTION	23
3.6. ARTICLE 6: EXISTING NUCLEAR POWER PLANTS	23
3.6.1. General	23
3.6.2. Existing Nuclear Power Plants in Argentina	23
3.6.3. Nuclear sector organization	23
3.6.4. Actions leading to safety improvements	24
3.6.4.1. Continuous execution activities	24
3.6.4.1.1. Documentation updating	24
3.6.4.1.2. Organization updating	25
3.6.4.1.3. Structures, Systems and Components inspection program	25
3.6.4.1.4. Periodic tests program	26
3.6.4.2. Specific improvement activities	26
3.6.4.2.1. CNA I improvements	26
3.6.4.2.1.1. Physical Protection System	26
3.6.4.2.1.2. New Emergency Power Supply System (EPS)	26
3.6.4.2.1.3. Other improvement activities	27
3.6.4.2.1.4. Lessons learned from national and international Operating Experience	27
3.6.4.2.2. CNE improvements	27
3.6.4.2.2.1. Physical Protection System	27

3.6.4.2.2.2. Lessons learned from national and international Operating Experience	27
3.6.5. NPPs life extension	28
3.6.5.1. CNE life extension	28
3.6.5.2. CNA I life extension	31
3.6.6. Opinion concerning the operation continuity of nuclear installations	31
3.6.7. Compliance with the obligations imposed by the Convention	32
3.7. ARTICLE 7: LEGISLATIVE AND REGULATORY FRAMEWORK	33
3.7.1. National Legislative Framework	33
3.7.2. Normative Framework	35
3.7.2.1. Introduction	35
3.7.2.2. Harmonization with IAEA Standards	36
3.7.2.3. Basic concepts	36
3.7.2.3.1. Deterministic and probabilistic aspects of the Regulatory Standards	37
3.7.2.3.2. Basic criteria of radiological and nuclear safety	37
3.7.3. Licensing System	37
3.7.3.1. General Aspects	37
3.7.3.2. Licensing Process	38
3.7.3.2.1. Nuclear power plant licensing	38
3.7.3.2.2. Periodic Safety Review and License Renewals	39
3.7.3.2.3. Nuclear power plant personnel licensing	39
3.7.3.3. Regulatory inspections and audits	40
3.7.3.4. Regulatory actions	41
3.7.4. Sanctions Regime	41
3.7.5. Compliance with the obligations imposed by the Convention	42
3.8. ARTICLE 8: REGULATORY BODY	43
3.8.1. Functions and competence of the Regulatory Body	43
3.8.2. ARN organizational structure	44
3.8.3. ARN human resources	45
3.8.3.1. Resources assigned to the nuclear power plants regulatory control	47
3.8.3.1.1. General aspects	47
3.8.3.1.2. Human Resources assigned to operation control	48
3.8.3.1.3. Human Resources assigned to construction and commissioning control	48
3.8.3.1.4. ARN personnel qualification	50
3.8.4. Quality Management System in the Regulatory Body	51
3.8.4.1. Process approach improvement	51
3.8.4.2. Document requirements	52
3.8.4.3. Satisfaction of stakeholders	53
3.8.4.4. Planning	53
3.8.4.5. Measurement, assessment and improvement	53
3.8.5. Communications with the stakeholders	54
3.8.6. Financial Resources	54
3.8.7. Relationship with other organizations	56
3.8.8. Compliance with the obligations imposed by the Convention	58
3.9. ARTICLE 9: RESPONSIBILITY OF THE LICENSEE HOLDER	59
3.9.1. Background	59
3.9.2. Licensee and Primary Responsible	59
3.9.3. Communication with the public	60
3.9.3.1. Communication campaigns	60
3.9.3.2. Relation with the media	60
3.9.3.3. Digital communication	60

3.9.3.4. Relation with the community	61
3.9.3.5. Plant tours	61
3.9.4. Regulatory control on the fulfilment of the Licencee Responsibilities	61
3.9.5. Compliance with the obligations imposed by the Convention	62
3.10. ARTICLE 10: PRIORITY TO SAFETY	63
3.10.1. General Principles	63
3.10.2. Specific actions	64
3.10.2.1. Safety policy	64
3.10.2.2. Safety culture and its development	65
3.10.2.3. Commitment to safety	65
3.10.2.4. Management attitudes towards safety	66
3.10.2.5. Personnel motivation	66
3.10.2.6. Safety culture regulatory assessment	66
3.10.2.7. Voluntary activities and good practices related to safety	67
3.10.3. Compliance with the obligations imposed by the Convention	67
3.11. ARTICLE 11: HUMAN AND FINANCIAL RESOURCES OF THE LICENSEE	68
3.11.1. Introduction	68
3.11.2. Electrical generation and economics	69
3.11.2.1. Total electrical generation period 2010 - 2012	69
3.11.2.2. Nuclear generation 2010, 2011, 2012	71
3.11.2.3. Human Resources	71
3.11.2.4. Current expenses 2010, 2011, 2012	72
3.11.3. Compliance with the obligations imposed by the Convention	73
3.12. ARTICLE 12: HUMAN FACTORS	74
3.12.1. System to detect, correct and prevent human errors	74
3.12.1.1. Human corrective actions in CNA I	74
3.12.1.2. Human corrective actions in CNA II	76
3.12.1.3. Human corrective actions in CNE	77
3.12.2. Management and institutional aspects	79
3.12.3. Human reliability analysis	80
3.12.4. Progress in human factors	80
3.12.5. Compliance with the obligations imposed by the Convention	80
3.13. ARTICLE 13: QUALITY ASSURANCE	81
3.13.1. Introduction	81
3.13.2. Quality Assurance Program of the Licensee	81
3.13.2.1. Quality supervision of external suppliers related to refurbishing and construction of NPPs	82
3.13.3. Compliance with the obligations imposed by the Convention	83
3.14. ARTICLE 14: ASSESSMENT AND VERIFICATION OF SAFETY	89
3.14.1. Introduction	89
3.14.2. Ageing	89
3.14.2.1. CNA I Reactor Pressure Vessel	91
3.14.2.2. CNE Pressure Tubes	92
3.14.2.2.1. Garter springs repositioning between Pressure Tubes and Calandria Tubes	92
3.14.2.2.2. Pressure Tube inspection and scrapping	93
3.14.2.2.3. Calandria Tube-Liquid Injection Shutdown System (LISS) nozzles contact	94
3.14.2.2.4. CNE Feeders	94
3.14.2.2.5. CNE Steam Generators	95
3.14.3. Safety assessment	98
3.14.3.1. Deterministic assessments	98
3.14.3.1.1. Analysis of loss of safety functions	98

3.14.3.1.1.1. Loss of Off Site Power (LOOP)	100
3.14.3.1.1.2. Station Black Out (SBO)	101
3.14.3.1.1.3. Loss of heat sinks	103
3.14.3.1.1.4. Loss of heat sink coincident with SBO	105
3.14.3.1.1.5. Regulatory actions	105
3.14.3.1.2. Accident Management and Severe Accident Management Program	105
3.14.3.1.2.1. CNA I	107
3.14.3.1.2.2. CNA II	109
3.14.3.1.2.3. CNE	109
3.14.3.1.2.4. Regulatory actions	111
3.14.3.1.2.5. Plant damage states characterization	111
3.14.3.1.2.6. Strategies proposal	111
3.14.3.1.2.6.1. CNA I	111
3.14.3.1.2.6.2. CNA II	113
3.14.3.1.2.6.3. CNE	114
3.14.3.1.2.7. Progress in the development of the accident progression model	115
3.14.3.1.2.8. Other issues related with severe accidents	115
3.14.3.1.2.9. Development of Instructions and Guidelines	115
3.14.3.1.3. CAREM Deterministic Safety Assessment	116
3.14.3.2. Probabilistic Safety Assessment	117
3.14.3.2.1. CNA I Probabilistic Safety Assessment applications	117
3.14.3.2.2. CNA II Probabilistic Safety Assessment applications	118
3.14.3.2.3. CNE Probabilistic Safety Assessment applications	119
3.14.3.2.4. CAREM Probabilistic Safety Assessment applications	120
3.14.3.3. Periodic Safety Review	120
3.14.4. Regulatory Plant Safety Performance Indicators	121
3.14.5. Compliance with the obligations imposed by the Convention	122
3.15. ARTICLE 15: RADIOLOGICAL PROTECTION	123
3.15.1. Introduction	123
3.15.2. Atucha I Nuclear Power Plant	123
3.15.2.1. Radioactive releases into the environment	123
3.15.2.2. Public exposure	124
3.15.3. Embalse Nuclear Power Plant	125
3.15.3.1. Radioactive releases into the environment	125
3.15.3.2. Public exposure	125
3.15.4. Occupational exposure	126
3.15.4.1. Dose limits to workers	126
3.15.4.1.1. Occupational dose at CNA I	127
3.15.4.1.2. Occupational dose at CNE	127
3.15.5. ALARA activities	127
3.15.5.1. CNA I Reactor Inspection Program	127
3.15.5.2. CNA I Steam Generators Inspection	129
3.15.5.3. CNE Steam Generators Inspection	129
3.15.6. Compliance with the Obligations Imposed by the Convention	130
3.16. ARTICLE 16: EMERGENCY PREPAREDNESS	131
3.16.1. Introduction	131
3.16.2. Emergency plans and programmes	132
3.16.2.1. ARN responsibilities related to nuclear emergencies	132
3.16.2.2. Requirements concerning emergency plans	133
3.16.2.3. Classification of emergency situations	134
3.16.2.4. On-site and off-site NPP emergency plans	135

3.16.2.5. National Nuclear Emergency Plan	136
3.16.2.6. Nuclear emergency plan at province scope	137
3.16.2.7. Nuclear emergency plan at local scope	137
3.16.2.8. Improvements concerning emergency preparedness in the ARN	137
3.16.2.9. Improvements concerning emergency preparedness in the NPPs	138
3.16.2.9.1. CNA I	138
3.16.2.9.2. CNA II	139
3.16.2.9.3. CNE	139
3.16.2.10. Actions taken for the improvement of the emergency exercises	140
3.16.3. Information of the public and neighbouring preparedness	142
3.16.3.1. Agreements with organizations concerning emergency plans	142
3.16.3.2. Bilateral agreement and the emergency conventions	143
3.16.3.3. Status of national and international agreements	143
3.16.4. Compliance with the obligations imposed by the Convention	143
3.17. ARTICLE 17: SITING	144
3.17.1. Introduction	144
3.17.2. Evaluation of site related factors	144
3.17.2.1. Normative aspects	144
3.17.2.2. Existing sites	145
3.17.2.3. Siting studies	145
3.17.2.3.1. Siting studies performed	145
3.17.2.3.2. Siting studies under execution	146
3.17.2.3.3. Site re-evaluation	146
3.17.2.3.3.1. CNA I and CNA II site re-evaluation	147
3.17.2.3.3.1.1. Earthquakes	148
3.17.2.3.3.1.2. Flooding / low water level	150
3.17.2.3.3.1.3. Other external hazards: tornadoes; wind loads; lightning and intense rains	151
3.17.2.3.3.1.4. Regulatory Actions	152
3.17.2.3.3.2. CNE site re-evaluation	152
3.17.2.3.3.2.1. Earthquakes	153
3.17.2.3.3.2.2. Flooding / low water level	154
3.17.2.3.3.2.3. Other External Events: Tornadoes; Wind loads; Lightning and Intense Rains	155
3.17.2.3.3.2.4. Regulatory Actions	156
3.17.3. Impact of the installations on individuals, society and environment	156
3.17.3.1. Environmental radiation monitoring	157
3.17.3.1.1. Atucha site	157
3.17.3.1.2. Embalse site	159
3.17.4. Compliance with the obligations imposed by the Convention	159
3.18. ARTICLE 18: DESIGN AND CONSTRUCTION	160
3.18.1. Introduction	160
3.18.2. Design and Construction	160
3.18.3. Compliance with Argentine Regulatory Standards: General Aspects	160
3.18.3.1. CNA I Nuclear Power Plant	161
3.18.3.1.1. Design improvements implementation	162
3.18.3.2. CNA II Nuclear Power Plant	162
3.18.3.2.1. CNA II components preservation	163
3.18.3.2.2. Critical technical issues established by ARN	164
3.18.3.2.2.1. Update the Quality System	165
3.18.3.2.2.2. Design Authority	165

3.18.3.2.2.3. Review of the licensing basic criteria	167
3.18.3.2.2.4. Review of safety aspects	167
3.18.3.2.3. Current design and engineering tasks	170
3.18.3.2.4. Current construction tasks	171
3.18.3.2.5. Design improvements implementation	171
3.18.3.3. CNE Nuclear Power Plant	172
3.18.3.3.1. Design improvements implementation	173
3.18.3.4. CAREM Reactor Prototype	173
3.18.3.4.1. Construction of CAREM prototype	174
3.18.3.4.2. Budget and human resources	175
3.18.3.4.3. Licenses and siting	175
3.18.3.4.4. Critical Technical Issues Established by ARN	176
3.18.3.4.4.1. Design Authority	176
3.18.3.4.4.2. Review of the Licensing Basic Criteria	176
3.18.3.4.4.3. Review of Safety Aspects	176
3.18.3.4.4.4. Design and Engineering Tasks	176
3.18.3.4.4.5. Construction Tasks	176
3.18.3.5. Fourth Nuclear Power Plant	176
3.18.4. Compliance with the Obligations Imposed by the Convention	176
3.19. ARTICLE 19: OPERATION	177
3.19.1. Introduction	177
3.19.2. Initial Authorization to Operate	178
3.19.2.1. CNA I - Initial Authorization to Operate	178
3.19.2.2. CNE - Initial Authorization to Operate	178
3.19.3. Operational Limits and Conditions, Maintenance, and Testing	178
3.19.3.1. CNA I - Operational Limits and Conditions, Maintenance, and Testing	178
3.19.3.2. CNE - Operational Limits and Conditions, Maintenance, and Testing	179
3.19.4. Inspection	179
3.19.4.1. CNA I - Inspections	180
3.19.4.1.1. In service Inspections	180
3.19.4.1.2. RPV Internal Components Surveillance Program	180
3.19.4.1.3. Fire Protection	180
3.19.4.2. CNE - Inspections	180
3.19.4.2.1. In service Inspections	180
3.19.5. Operational Procedures in Normal and Accidental Conditions	181
3.19.6. Engineering and Technical Support	181
3.19.7. Incidents Reporting	181
3.19.8. Operating Experience	182
3.19.8.1. Feedback from Local Operating Experience	182
3.19.8.2. Feedback of the Operating Experience of Other NPPs	184
3.19.8.3. Peer Reviews and Activities between the Licensee and WANO from 2007 to 2010	186
3.19.9. Radioactive Waste Management	189
3.19.9.1. Radioactive Waste Management Policy	189
3.19.9.2. Spent Fuel Management Policy	190
3.19.9.3. Spent Fuel Treatment and Storage at NPPs	190
3.19.9.3.1. Atucha I NPP	190
3.19.9.3.2. Embalse NPP	191
3.19.9.3.3. Atucha II NPP	191
3.19.9.4. Radioactive Waste Practices at NPPs	191
3.19.9.4.1. CNE life extension: waste management program	192
3.19.9.5. Research and Development Activities	192
3.19.9.6. Minimization of Radioactive Wastes	193

3.19.10. Compliance with the obligations imposed by the Convention	194
CHAPTER 4 – PLANNED ACTIVITIES TO IMPROVE SAFETY	195
4.1. REFURBISHMENT ACTIVITIES OF EMBALSE NPP	195
4.2. IMPROVEMENT TECHNICAL CAPACITY ON AGEING ASSESSMENT	195
4.3. IMPLEMENTATION of the IMPROVEMENTS ARISEN FROM STRESS TESTS	196
4.4. Internal Technical Support Missions	197
4.5. Use of own specific plant simulators as an additional tool for personnel training / retraining	197
ANNEX I – CONCLUSIONS ABOUT ARGENTINA DURING THE FIFTH REVIEW MEETING ON THE CONVENTION ON NUCLEAR SAFETY	
ANNEX II – ANSWER TO QUESTIONS OR COMMENTS – NATIONAL NUCLEAR SAFETY REPORT - 2010	
ANNEX III – MAIN TECHNICAL FEATURES OF THE ARGENTINE NUCLEAR POWER PLANTS IN OPERATION	
ANNEX IV – PRINCIPAL TECHNICAL CHARACTERISTICS OF ATUCHA II NUCLEAR POWER PLANT	
ANNEX V – PRINCIPAL TECHNICAL CHARACTERISTICS OF CAREM REACTOR PROTOTYPE	
ANNEX VI – EXAMPLES OF LESSON LEARNED AND CORRECTIVE ACTIONS RESULTING FROM NATIONAL AND INTERNATIONAL OPERATING EXPERIENCE AND EVENTS	
ANNEX VII – RESUME OF NA-SA QUALITY ASSURANCE GENERAL MANUAL CONTENT	
ANNEX VIII – REGULATORY REQUIREMENT RQ-NASA-38 COMPREHENSIVE SAFETY EVALUATION OF THE ARGENTINEAN NUCLEAR POWER PLANTS (STRESS TESTS)	

GLOSSARY

ABACC	<i>Agencia Brasileño-Argentina de Contabilidad y Control de Materiales Nucleares</i> (Argentine-Brazilian Agency for Accounting and Control of Nuclear Material)
AECL	Atomic Energy of Canada Limited
AMP	Ageing Management Program
AOO	Anticipated Operational Occurrences
AREVA	AREVA NP
ARN	<i>Autoridad Regulatoria Nuclear</i> (Nuclear Regulatory Authority)
ATWS	Anticipated Transients Without Scram
BDBA	Beyond Design Basis Accident
BP	Break Preclusion
BW	Babcock & Wilcox
CA	Condition Assessment
CAREM	<i>Central Argentina de Elementos Modulares</i> (Argentinean Nuclear Power Plant of Modular Elements)
CCE	Emergency Control Centre
CDF	Core Damage Frequency
CDS	Core Damage States
CECE	Control of Emergency External Centre
CFD	Computational Fluid Dynamic
CIAS	<i>Comité Interno Asesor de Seguridad</i> (Internal Safety Advisory Committee)
CICE	Control of Emergency Internal Centre
CIP	Public Information Centre
CNA I	<i>Central Nuclear Atucha I</i> (Atucha I Nuclear Power Plant)
CNA II	<i>Central Nuclear Atucha II</i> (Atucha II Nuclear Power Plant)
CNE	<i>Central Nuclear Embalse</i> (Embalse Nuclear Power Plant)
CNEA	<i>Comisión Nacional de Energía Atómica</i> (National Atomic Energy Commission)
COEM	Municipal Emergency Operative Centre
COG	CANDU Owners Group
CONUAR	Combustibles Nucleares Argentinos S.A.
CRM RD	<i>Comité de Revisión de Cambios de Diseño</i> (Relevant Design Changes Review Committee)
CRT	<i>Comité de Revisión Técnica</i> (Technical Review Committee)
CSS	Commission on Safety Standards
DBA	Design Basis Accidents
DA	Design Authority
DG	Diesel Generator
DSA	Deterministic Safety Assessment
DNPC	National Civil Protection Direction
ECC	Emergency Control Centre
ECCS	Emergency Core Cooling System
EMC2	Engineering Mechanics Corporation of Columbus
EMS	Environmental Management System
ENACE	Empresa Nacional de Energía S.A.
ENSI	Empresa Neuquina de Servicios de Ingeniería S.E.

EOL	End of Life
FAMOS	Fatigue Monitoring System
FE	Finite elements
FORO	Ibero-American Forum of Radiological and Nuclear Regulatory Agencies
FSAR	Final Safety Analysis Report
GIS	Geographical Information System
GRS	Gesellschaft für Anlagen-und Reaktorsicherheit
HCLPF	High Confidence of Low Probability of Failure
HPIP	Human Performance Improving Program
IAEA	International Atomic Energy Agency
ICR	Inappropriate Condition Report
ICRP	International Commission on Radiological Protection
IFMAP	Irradiated Fuel Management Advisory Programme
IMPSA	Industrias Metalúrgicas Pescarmona S.A.
INES	International Nuclear Event Scale
INSAG	International Nuclear Safety Group
INVAP	Investigaciones Aplicadas S.E.
IPERS	International Peer Review Service
IRRS	Integrated Regulatory Review Service
IRS	Incident Reporting System
ISI	In Service Inspection
IST	Industry Standard Toolset
IXP	International Exchange Program
JOEN	Nuclear Emergency Operative Chief
KWU	Kraftwerk Union
LAC	Local Air Coolers
LBB	Leak Before Break
LISS	Liquid Injection Shutdown System
LOCA	Loss of Coolant Accident
LTO	Long Term Operation
MCCI	Molten Core-Concrete Interaction
MDG	Mobile Diesel generator
NA-SA	Nucleoeléctrica Argentina S.A. (Licensee)
NDE	Non Destructive Examinations
NERS	ARN-Nuclear Emergency Response System
NPP	Nuclear Power Plant
NSGC	Nuclear Security Guidelines Committee
NUSSC	Nuclear Safety Standards Committee
OEP	Operating Experience Program
OPEX	Operating Experience
OSART	Operational Safety Review Team
PARs	Passive Autocatalytic Recombiners
PAZ	Precautionary Action Zone
PGA	Peak Ground Acceleration
PHTS	Primary Heat Transport System
PHWR	Pressurized Heavy Water Reactor
PLEX	Plant Life Extension

PLiM	Plant Life Management
PMH	Probable Maximum High Water Level
PSA	Probabilistic Safety Assessment
PSR	Periodic Safety Review
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
RWO	Risk Work Order
SAGSI	Standing Advisory Group on Safeguards Implementation
SAMP	Severe Accident Management Program
SAR	Safety Analysis Report
SAT	Systematic Approach to Training
SBO	Station Black Out
SC	Safety Culture
SEDA	Dose Assessment System Accident
SL	Seismic Level
SCK/CEN	Studiecentrum voor Kernenergie
SIEMENS	SIEMENS Kraftwerk Union AG
SIEN	Emergency Response System
SMA	Seismic Margin Assessment
SMR	Small and Medium Reactor
SPI	Safety Performance Indicators
SSA	Safety Seismic Assessment
SSC	Structures, Systems and Components
SSEL	Safe Shutdown Equipment List
TM	Technical Meeting
TRANSSC	Transport Safety Standards Committee
TSP	Total Suspended Particulate Matter
TÜV	<i>Technischer Überwachungs Verein, Baden</i> (German Inspection Organization)
UCE	Education and Training Unit (<i>Unidad de Capacitación y Entrenamiento</i>)
UHS	Uniform Hazard Spectra
UNIPi	University of Pisa
UNSCEAR	United Nations Scientific Committee on the Effects of Atomic Radiation
UPZ	Urgent Protective Zone
USNRC	US Nuclear Regulatory Commission
UTN-BA	<i>Universidad Tecnológica Nacional - Buenos Aires</i> (National Technological University)
WANO	World Association of Nuclear Operators
WASSC	Waste Safety Standards Committee
WHO	World Health Organization
WP	Work Permit

INTRODUCTION

This Sixth 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.

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

The present report also takes into account the observations and discussions made during the Fifth Review Meeting, and *include the actions taken in the light of the Fukushima Daiichi accident.* The conclusions made in the previous fifth review meeting about the compliance by Argentina with the obligations of the Convention are included as Annex I. The questions and answers originated at the fifth review meeting are included as Annex II.

With this same purpose, Annex III shows some design characteristics of NPPs in operation Atucha I (CNA I) and Embalse (CNE); Annex IV describes the design characteristics of the Atucha II NPP under commissioning (CNA II), Annex V shows information relative to the CAREM reactor, small power prototype whose construction is anticipated in the Argentine nuclear program, Annex VI shows examples of lesson learned and corrective actions resulting from national and international operating experience and events, Annex VII is a summary of Licensee's Quality Assurance programme content and, *Annex VIII presents the regulatory requirement for carry out stress tests to the Argentinean NPPs in response to the Fukushima Daiichi accident.*

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 Act N° 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 Sixth National Nuclear Safety Report describes the actions the Argentine Republic has carried out since the Fifth Nuclear Safety Report was issued (March 2010) until 2013, 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 issued licenses and other regulatory decisions. The corresponding information is described in the analysis of each of the Convention Articles constituting this Report.

Argentina has two NPPs in operation, CNA I and CNE, which initiated their commercial operation in 1974 and 1984 respectively. Their corresponding net electric powers are 335 MW and 600 MW, which represents some **3.23%** of the installed electric power. Both NPPs supply about **4.83%** of the total electric power generated *in 2012*.

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

CNE is located in the Province of Córdoba, approximately 110 km to the south of Córdoba City (state capital) 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.

Atucha II NPP (CNA II) is under the **commissioning stage** and is located in the same CNA I site (Atucha site). Its reactor will also be of the PHWR type with a **RPV**, loaded with natural uranium fuel, moderated and cooled with heavy water, and with a net electric power of 693 MW.

The CAREM reactor prototype (CAREM) is an Argentine design in an advanced development stage, with a planned electric power of 25 MW, and will be constructed also in the Atucha site. Its principal design characteristics are integrated primary system, self-pressurization, and Passive safety systems.

For a Fourth NPP, feasibility studies are in progress to define the type of reactor and the appropriate siting.

1.2. NATIONAL POLICY IN THE NUCLEAR FIELD

Due to its special characteristics, the activities related to the use of nuclear energy for peaceful purposes needs to be subject to national (or federal) jurisdiction and regulated as an organic and indivisible system. For this reason the Argentinean National Congress is empowered to establish the laws concerning the subject, through Section 75 paragraphs 18 and 32 of the Constitution.

Within this context, Act Nº 24.804, 1997 or "National Law of the Nuclear Activity" is the legal framework for the peaceful uses of nuclear energy. Article 1 of the mentioned Act, states that concerning nuclear matters the State will establish the policy and perform the functions of research and development through the National Atomic Energy Commission (CNEA) and of regulation and control by the Nuclear Regulatory Authority (ARN).

In the case of nuclear power generation, Article 4 of Decree Nº 1.540, 1994, sets that a State Company named Nucleoeléctrica Argentina S.A. (NA-SA) is in charge of the generation activities of CNA I and CNE and of the construction, commissioning and operation of CNA II.

Moreover, the mentioned law sets that any nuclear activity either productive or concerning research and development, that may be commercially organised, can be carried out by both the State and the private sector.

Article 1 of the Act states that in the implementation of the nuclear policy the obligations assumed by Argentina under the Treaty for the Prohibition of Nuclear Weapons in Latin America and the Caribbean (Tlatelolco's Treaty), the Treaty on Non-Proliferation of Nuclear Weapons, the Agreement between the Argentina Republic, the Federative Republic of Brazil, the Brazilian-Argentine Agency for Accounting and Control of Nuclear Materials, and the International Atomic Energy Agency for the Application of Safeguards, as well as the assumed commitments under the membership of the Nuclear Suppliers Group, and the National System of Control of Sensitive Exports, must be strictly observed.

1.3. NATIONAL PROGRAM CORRESPONDING TO NUCLEAR INSTALLATIONS

The construction and commissioning of the CAREM (Decree of the National Executive Power Nº 1.107/2006) for nuclear energy generation was declared of national interest, being the execution of the work necessary for the construction and commissioning of this prototype under the technical responsibility of CNEA.

The National Government continues promoting nuclear activities in the country. Within this framework, the National Congress, through Act 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.

Act 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.

NA-SA is analysing different technological alternatives regarding the possible construction of Argentinean's Fourth NPP, taking into account mainly the aspects related to design, transfer of technology and the possibility of local suppliers of goods and services. *During 2010 and 2011, meetings with potential suppliers (one or two units) took place. These meetings included heavy and light water reactors suppliers. A group of suppliers were already prequalified.*

Activities related with the CNE's Life Extension Project, to extend the plant life for another 25 years of full power operation as well as to increase the electrical power in about 35 MW are in an advanced stage. Contracts with CANDU Energy (ex AECL) and other suppliers were signed in 2011.

1.4. SUMMARY OF THE MAIN SUBJECTS CONTAINED IN THE REPORT

This Sixth 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 Reports Under the Convention on Nuclear Safety, the most significant conclusions introduced during the Fifth Review Meeting in 2011, *as well as the First and the Second Extraordinary Meetings of the Contracting Parties of the Convention on Nuclear Safety which were carried out in*

2011 and 2012 respectively. This means that the Report has been organized according to the Articles of the Convention and the contents as indicated in the above-mentioned Guidelines.

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

Chapter 2 of this report contains follow-up information on issues raised or requested by other countries at the Fifth 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, and 5th 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 result as a consequence of operational experience or in response to regulatory requirements.
- Article 7 presents the legal and regulatory structure that rules nuclear safety. It also analyses the criteria supporting the normative framework and the NPP licensing and regulatory control system.
- Article 8 shows functions, responsibilities and organizational structure of the Regulatory Body, human and economic resources, personnel qualification and its relationships with other official and private organizations.
- Article 9 describes the Licensee's responsibilities and the controls required to verify the compliance with such responsibilities.
- Article 10 analyses the policies and the priority on nuclear safety established by the Regulatory Body as well as the Licensee.
- Article 11 deals with functions, responsibilities and the structure of the Licensee as well as the human and economic resources and the personnel qualification.
- Article 12 analyses the systems required to detect prevent and correct human errors.
- Article 13 shows the Licensee quality assurance program in the design, construction and operation of NPPs.
- Article 14 deals with deterministic and probabilistic safety assessments performed by the Licensee, and the safety assessments, evaluations and verifications performed by the Regulatory Body at every stage of the nuclear installation's lifetime.
- Article 15 describes the radiological safety criteria used, the existing rules on the subject, the authorised discharge limits, dose evaluations to workers of NPPs and to the public, and ALARA applications.
- Article 16 presents the laws, regulations and requirements existing in the country and their implementation in case of a radiological emergency at a NPP. It analyses the actions to be taken inside and outside each plant, by all the intervening organizations, with special emphasis in training exercises in the application of the emergency plan.
- Article 17 summarises the studies related to NPPs siting and site re-evaluation studies.
- Article 18 analyses the design and construction of NPPs and their compliance with the Argentine standards as well as application of good international practices, principles of defence in depth, diversity and redundancy. *Besides, has been included the design improvements and the plant modifications which were already implemented or that are foreseen to be implemented to prevent BDBAs or to mitigate their radiological consequences.*
- Article 19 deals with the mandatory documentation for NPPs operation, the technical support given to the installations, the operational limits and conditions, the maintenance activities and tests, the feedback mechanism of operational experience (*OPEX*), fire protection and relevant events communication, peer-review activities, and radioactive waste management.

Finally, Chapter 4 of this Report addresses the planned activities to improve safety, such as:

- *Refurbishment activities of Embalse NPP.*
- *Improvement technical capacity on ageing assessment.*

- *Implementation of the improvements arisen from stress tests.*
- *Internal technical support missions.*
- *Use of own specific plant simulators as an additional tool for personnel training / retraining.*

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 and the Nuclear Safety Convention (NSC), as well as maximize the benefit of the lessons learned from the mentioned accident.

Additionally, member countries 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 implemented by the Western European Nuclear Regulators Association (WENRA), in order to detect weaknesses to face in case of extreme events and to propose the corresponding improvements. The European experience was incorporated in order to improve and adapt the FORO's one to the technology of the NPPs existing in the region.

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 loss of off-site and onsite power (station black out, SBO) and the loss of ultimate heat sink for a long term, as well as the capacity to manage such accidents.

In order to achieve this goal, consensus was reached between the FORO member countries regarding the stress tests content and scope, taking into account the specific circumstances of the FORO members, so that each Regulatory Body had 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).

The Argentine Regulatory Body formalized this stress test by a regulatory requirement (See Annex VIII) to the Licensee of CNA I, CNA II and CNE. This requirement consists of a reassessment of the NPPs safety margins assuming the occurrence of a sequential loss of the lines of defence in depth caused by extreme initiating events and includes:

- *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.*

Many opportunities for improvements have been identified but, in no case were showed weaknesses or critical situations (cliff edge) which make necessary to take urgent actions

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 (Argentina, Brazil, Chile, Cuba, Mexico, Peru, Spain, and Uruguay). The review process allowed to verify that the assessment made by Argentina demonstrates the existence of suitable margins to fulfil the safety functions in case of the occurrence of the extreme accidental situations proposed.

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 NSC 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 latter reaching the year 2016. The FORO members decided to perform two follow-up technical meetings to verify the implementation of such improvements.

Moreover, in the Argentine National Nuclear Safety Report to the 2nd NSC Extraordinary Meeting, were included a description of the activities that have been completed and are under development, as well as the activities foreseen, for the Argentine NPPs in response to the lesson learned of the Fukushima accident.. The above mentioned related activities and the planned improvements, some of which have already been implemented, are listed below:

1.4.1.1. External events (see sections 3.17.2.3.3.1.).

Earthquakes (see section 3.17.2.3.3.1.1. and 3.17.2.3.3.2.1.)

- Safety seismic assessment (SSA) based on the corresponding updated seismic hazard (CNA I / CNA II).*
- Easy fixes (CNA I).*
- Installation of new seismic instrumentation (CNA I).*
- Seismically induced internal flooding (CNA I / CNE).*
- Plant walkdown to verify the seismic capability (CNA II).*
- As a Regulatory Body proactive decisions it was decided to perform an independent updated assessment of the seismic hazard. Besides, a comparison between the results of this study and the corresponding one addressed by the licensee was requested. (CNA I / CNA II).*
- Geological investigations in order to fully characterize the seismotectonic setting of the Embalse region (CNE).*
- Comprehensive review of the site seismic hazard to be used in the SSA through a SMA based on PSA (CNE).*
- Addition of a seismically qualified rupture disc to the calandria vault (CNE).*
- Seismic verification of the silos of the Spent fuel dry storage system (ASECQ) (CNE).*

Flooding / low water level (see sections 3.17.2.3.3.1.2. and 3.17.2.3.3.2.2.)

- Strategy for refilling the pools through a water supply system pump (UJ) (CNA I).
- New hydrologic and hydraulic study (CNA I / CNA II).
- Procedure for systematizing the outage of the plant in case of extreme low-water-levels (CNA I).
- Installation of an additional (fourth) pump to the river Water Cooling Ensured System (UK) (CNA I).
- Increase in time water inventory reposition to the SGs independent of secured river water (CNA I).
- Procedures for cooling the plant via the SGs in case of SBO and loss of the assured service water cooling system (CNA II).
- Re-evaluation of the consequences of probable earthquakes on the existing dam located downstream (CNE).
- Air compressor driven by a DG located in the turbine building (CNE).
- Secondary control room gateway reinforcement (CNE).
- Relocation of the site where is located the centre of the plant lighting system (CNE).

Other external hazards (see section 3.17.2.3.3.1.3. and 3.17.2.3.3.2.3.)

- Analysis of other possible external sources of flooding different from Parana River (CNA I / CNA II).
- Re-evaluation of the plant original conditions of object impacts thrown by tornadoes (CNA I).
- Re-evaluation of the risk of tornadoes (CNA I / CNA II / CNE).

1.4.1.2. Design issues / Analysis of loss of safety functions (see section 3.14.3.1.1.):

Loss of Off Site Power (LOOP) (see section 3.14.3.1.1.1.)

- Restoration of external power supply (CNA I).
- Passive components control (CNA I).
- New Emergency Power Supply System (EPS) (CNA I).
- EPS additional modification (CNA I).
- Electrical interconnection between normal bars of CNA I / CNA II.
- Assessment of the external electrical supply lines availability (CNA I).
- Restoration of external power supply (CNA II).
- Assessment of the external electrical supply lines availability (CNA II).
- Review and improvement of emergency procedures (CNA II).
- External power supply protection devices (CNE).
- Upgrade of the EPS system (CNE).
- Replacement of the Class III DGs (CNE).

Station Black Out (SBO) (see section 3.14.3.1.1.2.)

- Alternative power sources –MDGs- (CNA I / CNA II / CNE).
- Preventive strategies to avoid core damage (CNA I).
- Fuel elements integrity assessment (CNA I / CNA II / CNE).
- Extension of the batteries availability (CNA I / CNA II / CNE).
- Connection among the DGs of CNA I and CNA II (CNA I / CNA II).
- Abnormal event procedure (CNE).

Loss of heat sinks (see section 3.14.3.1.1.3.)

- Alternative water sources:
 - CNA I: process water refilling of the spent fuel storage pools by using an alternative reservoir (install an independent pump) and; replenishment of water inventory to the SGs through the SHS system.

- o CNA II: alternative water reservoir that allows maintaining the water supply to heat removal through the SGs and cooling the spent fuel storage pool and; additional system to water replenishment of the spent fuel storage pools
- o CNE: installation of a 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 and; additional fire truck containing 17,000 litres of water.
- Modifications to the emergency water supply (EWS) system (CNE).
- Abnormal event procedure (CNE).

Loss of heat sinks coincident with SBO (see section 3.14.3.1.1.4.)

- Instrumentation & Control (I&C) (CNE): I&C system in the secondary control room to allow monitoring the level and water temperature of the spent fuel storage pool and; improvements in I&C used to measure critical parameters.

1.4.1.3. Severe accident management and recovery –on site– (see sections 3.14.3.1.2.1., 3.14.3.1.2.2. and 3.14.3.1.2.3.)

- Plant Model Using MELCOR Code (CNA I).
- Severe Accident Management Guidelines (SAMG) (CNA I / CNE).
- Implementation of WANO SOER 2011-2 recommendations (CNA I).
- Procedure for operation in perturbations and accidents –abnormal events– (CNA I / CNE).
- Passive auto-catalytic recombiners (PARs) (CNA I / CNA II / CNE).
- Dose rate remote measurement system (CNA I / CNA II).
- Strategies to reduce the containment pressure during severe accidents (CNA I).
- Cooling of the RPV external side (CNA I / CNA II).
- Review and improvement of accident management procedure (CNA I).
- Severe accident management program (CNA I / CNA II).
- Venting filtered containment system (CNA I / CNA II / CNE).
- I & C (CNA I / CNA II / CNE).
- Alternative cooling mode of the DGs (CNA I / CNA II).
- Disconnection of electrical loads (CNA I / CNA II).
- Increasing DGs autonomy (CNA II).
- Switchgear building ventilation (CNA II).

1.4.1.4. Emergency preparedness and response and post-accident management –off site– (see section 3.16.2.9.)

- Environmental monitor net around the plant (CNA I / CNA II / CNE).
- Source term verification and update (CNA I / CNA II / CNE).
- New fix and mobile satellite phones (CNA I / CNA II / CNE).
- Foam and air system against fire (cannon for long distance action) (CNA I).
- Revision of current procedures applicable to the strategy of emergency management (CNA I).
- Modification of the Procedure PS-101 “Conformation and function of the control of emergency internal centre – CICE” (CNA I).
- Conditioning of a municipal emergency control centre far from the planning zone of the nuclear emergency (CNA I / CNA II / CNE).
- Emergency exercises extended in time (long term) (CNA I / CNA II / CNE).
- Relocation of the Emergency Control Internal Centre (CICE) (CNA I / CNA II).
- Analysis of different facilities to retain the water used in decontamination tasks (CNA I).
- New building to house a meeting point with a capacity of 1,300 persons (CNA I).
- Re-evaluation of all the current procedures applicable in the emergency management strategy (CNE).

- *Provision of portable electric generators with light columns and MDGs (CNE).*
- *Electric panel installed outside the service building with connection facilities to the emergency light equipment (CNE).*
- *Review and improvement of the procedures applied to the co-ordination and reception of supplies, equipment and additional personnel (CNE).*
- *Small trenches with plastic membranes (CNE).*
- *Modification and improvement of the Emergency Internal Control Centre (CICE) building (CNE).*
- *Installation of an air recirculation filters system in the secondary control room (SCR) (CNE).*

The general conclusion resulting from the stress tests performed by the Licensee and the corresponding assessment carried out by the Argentinean Regulatory Body is that were verified the need for some regulatory actions but not relevant weaknesses that require urgent actions as well as 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 Regulatory Body. Therefore, it was issued a regulatory requirement asking improvements and modifications referred to the issues above listed to be implemented in each NPP. This regulatory requirement includes short, medium and long term actions; the latter reaching the year 2016.

1.4.2. ANNEXES

Additional information is included in the following Annexes:

Annex I: Conclusions about Argentina during the Fifth Review Meeting on the Convention on Nuclear Safety.

Annex II: Answer to Questions or Comments - National Nuclear Safety Report - 2010.

Annex III: Main Technical Features of the Argentine Nuclear Power Plants in Operation.

Annex IV: Principal Technical Characteristics of Atucha II Nuclear Power Plant.

Annex V: Principal Technical Characteristics of CAREM Reactor Prototype.

Annex VI: Examples of Lesson Learned and Corrective Actions Resulting from National and International Operating Experience and Events.

Annex VII: Resume of NA-SA Quality Assurance General Manual Content.

Annex VIII: *Regulatory Requirement RQ-NASA-38 Comprehensive Safety Evaluation of the Argentinean Nuclear Power Plants (Stress Tests)*

CHAPTER 2

FOLLOW-UP FROM THE FIFTH REVIEW MEETING

At the Fifth Review Meeting in 2011, the Rapporteur resumed the challenges and planned measures to improve safety for Argentina on specific topics to follow-up on this Sixth Report. These topics are discussed in the following sections.

2.1. THE SPENT FUEL STORAGE AT ATUCHA I NPP

The available positions in the spent fuel storage pools at CNA I will be completed in February 2015 while the EOL plant design will be get by December 2017, thus a new option must be available for the spent fuel storage.

Therefore, NA-SA with the CNEA assistance has studied different alternatives for interim dry storage and a project has been developed. This project consists of a new building, where vertical underground silos will be located, constructed as an extension of the spent fuel storage building and within the plant controlled area. Besides, this project includes the construction of a transference flask (shield for the spent fuel) for the spent fuel transportation from the spent fuel storage pool (wet area) to the silos (dry area).

In January 2013 the interference settlement stage in the site of the new spent fuel storage building was finalized. The basic engineering, which is CNEA's responsibility, is still under execution.

Subsequently, the stage of civil works began with the construction of a net for sustaining soil, excavation and pit foundation and is currently on-going. The works responsibility was delegated to the Construction Unit of CNA II and the Fourth NPP Construction Group on the following topics:

- Execution the detailed engineering Project.*
- Execution of civil works.*
- Execution the electromechanical erection.*
- Commissioning.*

Despite this, the schedule is too tightened for that the facility be operational by February 2015. For this reason and in order do not to affect the operation of CNA I, the Construction Unit of CNA II is working on the following alternatives:

- Releasing the places currently occupied by internal components of the reactor located in the spent fuel storage pools. During the 2013 scheduled outage the plant began to release such places in such a way that would allow about four extra months of operation.*
- Transfer spent fuel elements from the Unit I to the Unit II: By the end 2012 the conceptual engineering project to transfer 620 spent fuel elements from Unit I to Unit II was began with the firm AREVA-Transnuclear participation. At present, a negotiation is under way with the same company to carry out the detailed engineering and Preliminary Safety Report for this project.*

The Licensing process of equipment and transfer operation of spent fuels will begin when this documentation is provided. Basically, it is foreseen that the fabrication of casks and other devices necessary for the transfer operation should be available in September 2014 in order to begin the process of sending spent fuels to the CNA II spent fuel storage pools.

With the implementation of the proposed transfer of the spent fuels from CNA I to CNA II, CNA I will reach the EOL design. At the same time, this solution serves as an alternative until the spent fuels dry storage project is finalized.

Furthermore, IAEA developed the Irradiated Fuel Management Advisory Programme (IFMAP); the first IFMAP mission was conducted, in March 2012, to the Spent Fuel Dry Storage Project of CNA I. An international experts' team reviewed the project's technical and planning documentation, suggested improvements to the conceptual engineering, and issued a report covering technical and organizational issues.

The report included recommendations for interaction with the Regulatory Body and technical improvements such as an enhanced drying procedure. Backup options in case of delays, including the temporary use of the spent fuel storage pools of CNA II, were also discussed.

2.2. ADEQUACY OF TRAINING FOR STAFF FOR THE REGULATORY BODY AND KNOWLEDGE RETENTION AS WELL AS FOR THE LICENSEE

REGULATORY BODY

The ARN is recognized by the IAEA as the organization in charge of the Regional Training Centre (RTC) in Latin America and The Caribbean for Nuclear, Radiation, Transport and Waste Safety and based on more than 30 years of successfully conducting Postgraduate Educational Courses (PGEC) on Radiation Protection and Safety. A training course on Radiation Protection at technical level is also available.

ARN continues involving retired expert personnel, as advisers emeritus; in the areas that requires their expertise. Fourteen emeritus were nominated in the last five years.

In 2010 the ARN Board of Directors created the Education and Training Unit (UCE), to perform the following activities:

- Fulfil the ARN's obligations and develop the corresponding activities as IAEA RTC headquarters.*
- Propose strategies and action courses to the Board of Directors for the education and training of the ARN staff on topics related to the ARN specific functions as well as to manage the resulting activities.*
- Ensure the institutional knowledge management.*

The activities of the UCE have been focused to its objective of maintaining and enhancing the quality of their PGEC, where the ARN has formed more than 1000 professionals, of which 45% were Argentines.

Other activities managed by the UCE, during the 2010 / 2012 period, related to the formation of regulatory staff were:

- The development of 5 courses of radiological protection for technical level personnel (1 course in 2010 and 2 courses both in 2011 and 2012) with 240 hours duration each were aimed to the undergraduate level staff. In these courses a total of 131 participants have participated, of which 29 participants (22%) are part of the ARN technical staff.*
- The organization of specific internal courses that have been demanded by different ARN's Divisions and Departments.*
- The arrangements to continue with the seminar cycles imparted by expert staff which are near the retirement.*
- The analysis of the external refinement programs applied to senior personnel.*

ARN staff training on safety issues, in particular newcomers is being addressed through their attendance to the PGEC courses.

On a case by case basis, intensive on the job training, specialized short-medium training courses (domestic and foreign organizations), workshops, fellowships, knowledge management activities and the assignment of tasks under supervision by senior staff following a graded approach with respect to the complexity of the duties, are also performed. Collaborative activities through contracts or cooperation agreements with domestic and foreign organizations to train and cover ARN's new challenges were also performed. Subsequently, those professionals assigned to NPPs inspections and other licensing and regulatory activities receive a wide and complete specific training in NPPs.

ARN's and Licensee's experts take part in safety related activities in organizations like ICRP, UNSCEAR and IAEA (e.g. CSS, INSAG, IRS, NSGC, NUSC, RASSC, TRASSC, WASSC consultant and technical meetings). They also participate in OSART, IRRS and expert missions in different foreign for a. Licensee's experts participate in WANO activities in other plants in the world (see section 3.19.8.3.).

LICENSEE

CNA I - CNA II

The training in CNA I was increased during the last years. The time devoted to the training or re-training during the last three years is the following: 4,264 hours in 2010, 4,620 hours in 2011 and, 6,351 hours in 2012.

Historically only teams of Operation and Safety received retraining. In 2011 a new committee was formed in order to evaluate training plans for all sectors, including Maintenance Engineering and Production. In these sense senior personnel train the rest of the personnel in each sector.

It has to be taken into account that the training requirements for CNA II positions have been defined in order to have specific training programs for each one of them. Additionally, There was a specific contract for technical assistance from TECNATOM to develop and implement a Systematic Approach to Training (SAT) for the CNA II organization chart positions.

CNA II personnel are using the new Full Scope simulator constructed by TECNATOM. CNA I personnel continue being trained in the Angra II, Brazil, full scope simulator, as it does since 1997.

Additionally, for both units CNA I and CNA II operation personnel uses the two Graphic Interactive Simulators, one for each plant

CNE

The time devoted to the training or re-training indicates that during 2012 were executed 43,655 hours more than those which were planned (36,980 hours). In 2013 from 58,920 hours planned 46,163 hours have been already executed.

In 2013, operations shifts will be trained using the new full scope simulator located at the plant. Before, this training was performed in the full scope simulator of Gentilly II NPP, Canada.

New plant procedures for the use of SAT (Systematic Approach to Training) were developed and the first training using this approach was done according to what it appeared in the evaluation stage.

2.3. LICENSEE HUMAN PERFORMANCE PROGRAMS

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

For more details see article 3.12.

2.4. NPPs' REFURBISHMENT AND CONSTRUCTION PROGRAMS

CNE Life Extension

NA-SA, with the CNE's designer support (CANDU Energy -ex AECL-), has performed a CNE life extension feasibility study in order to establish the scope, schedule and the necessary project investments. Therefore, NA-SA has 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 long term operation (extended period of 25 – 30 year). The PLEX program is divided into three phases:

- Phase 1: Project Definition.
- Phase 2: Project Engineering and material supply (preparation).
- Phase 3: Project Implementation.

The Regulatory Body required to the Licensee, as a condition for licensing the CNE LTO, to carry out a Periodic Safety Review (PSR), following the IAEA NS-G-2.1 (see section 3.14.3.3.). Besides, an update of the mandatory documentation including the FSAR was required.

The licensing process has also followed the requirements established by the Argentine Regulatory Standards (AR standards), the applied Canadian Standards (Canadian Nuclear Safety Commission)

due to Canada is the CANDU technology origin country, as well as IAEA standards and international practices.

The scope and status of the tasks that are being undertaken in the framework of the PLEX of CNE are detailed in article 3.6.

CNA I Life Extension

The Licensee has presented to the Regulatory Body a preliminary report on the feasibility of conducting a life extension for CNA I which includes mainly a current plant state assessment and the tasks that have been made since its commissioning which resulted in plant safety improvements. Besides, the organization to conduct the PLEX activities as well as the tasks proposed for the different stages were presented. Additionally, the Licensee is analysing an alternative for managing the life extension for subsequent periods of 5 years. The Regulatory Body is evaluating the above mentioned preliminary report.

2.5. INSTALL FULL SCOPE TRAINING SIMULATORS

The full scope training simulators will support personnel training and retraining, by meeting the requirements set forth in the standards AR 0.11.3. and IAEA NS-G-2.8 - "Recruitment, Qualification and Training of Personnel for Nuclear Power Plants".

CNA I

A new interactive reduced scope windows basis graphic simulator was acquired to be added to the existing one as an additional tool for personnel training / retraining.

CNA II

A full scope simulator and an interactive graphic simulator were commissioned to the enterprise TECNATOM and it was performed with NA-SA personnel participation. Both simulators have 120 plant systems incorporated. The full scope simulator is being used for operating shifts training since June 2013. Models are foreseen to be updated with the latest engineering changes and commissioning data.

CNE

A Full Scope Simulator developed by the enterprise L3-MAPPs was acquired, This simulator is similar to those installed in similar CANDU NPPs around the world and, will be the tool used for training and retraining of operating personnel on normal operation and accident conditions within the design basis, during the refurbishment period and the second operating cycle (extended plant operation).

2.6. EXTERNAL HAZARDS ASSESSMENTS INCLUDING FLOODING AND SEISMIC RE-EVALUATION

Site Re-Evaluation

The selection of CNA I, CNA II and CNE sites, were performed, according to the criteria and the information available, during the sixties and the early seventies. Nowadays, there are more accurate tools to define the site 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 safety plant condition:

- Existence of new and more reliable data and information for assessing the occurrence of external event hazards, in comparison with those available at the time of the original design.
- Availability of methods and criteria consolidated through their use in different regions (i.e. different natural 'environments') and countries (i.e. different regulatory and practice 'environments') allowing to reach certain consensus in the international practice about what to do and how to do it.
- A more balanced situation between the regulatory requirements and the industry practices, through the use of more realistic and integrated criteria, with a trend to reduce the use of excessive conservatism.

- Changes in both the real plant configuration and the present conditions in the nearby region; as well as those modifications introduced to structures, systems and equipment to improve their performance.

The design parameters corresponding to earthquakes, extreme meteorological phenomena and man induced events were determined according to the region and the plants site specific conditions.

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 NPP's 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 with the purpose of apply the corresponding lessons learned, the Regulatory Body requested to perform a stress test to each Argentinean NPP (See Annex VIII) 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 were showed weaknesses or critical situations (cliff edge effects) which make necessary to take urgent actions.

The improvements and modifications proposed by the NPPs Licensees included an implementation schedule composed by short, medium and long term actions; the latter reaching the year 2016. Finally, as result of the stress tests, it was issued a regulatory requirement containing numerous improvements and modifications to be implemented by the Argentinean NPPs, some of which have already been implemented.

More details about external events assessments including flooding and seismic re-evaluations are included in article 3.17.

2.7. SEVERE ACCIDENT MANAGEMENT

As was mentioned in previous CNS Reports, the ARN required the Licensee to develop a Severe Accident Management Program (SAMP) for both plants in operation, starting with CNA I. The initial activities were (see 2004 CNS Report):

- 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 (see 2007 CNS Report), as follows:

- Plant damage states characterization including deterministic studies related to new scenarios (or extending the existing ones) with RELAP code up to core uncovered initiation.
- Strategies proposal (preventive strategies for the time).
- Small LOCA plus failure of low pressure emergency water injection.
- Small LOCA plus moderator pumps failure which delays low pressure emergency water injection due to depressurization delay.
- Blackout (SBO).
 - o Unavailability of the second heat sink pumps or lack of sufficient inventory for different accidental scenarios.
 - o 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 CNE SAMP development has been considered integrating the tasks related to the life extension project (see 2010 CNS Report). 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 a consequence of the Fukushima accident and with the purpose of apply the corresponding lessons learned, the Regulatory Body requested perform a stress test to each Argentinean NPP (See Annex VIII) 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 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 alternating 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 were showed weaknesses which make necessary to take urgent actions.

The improvements and modifications proposed by the NPPs Licensees included an implementation schedule composed by short, medium and long term actions; the latter reaching the year 2016 which were considered acceptable by the Regulatory Body. Finally, as result of the stress tests, it was issued a regulatory requirement containing the improvements and modifications to be implemented by the Argentinean NPPs, some of which have already been implemented.

More details are in article 3.14.

2.8. USE OF INTERNATIONAL REVIEWS

WANO PEER REVIEWS activities

The Licensee is a WANO member (from 1988), at first as part of CNEA and then as associated member when NA-SA was ratified in Paris (1995).

CNA I and CNE have received WANO Peer Review Missions (PR) and the corresponding Monitoring Mission (MS). See details in Table I.

In order to improve particular areas of CNA I and CNE, NA-SA requested to WANO assistance through Technical Support Missions (TSM), as well as, regular and monitoring support visits every four months on the plants sites, (see details in Table 1). These areas are:

- Safety Culture
- Operating Experience
- Industrial Safety
- Tasks observation
- Radiological Protection

Table 1

PERIOD	HOST	ACTIVITY
February 11-15, 2005	CNA I and CNE	TSM
June 2005	CNA I and CNE	Final report
December 12 -15, 2005	CNA I	PR Pre-visit
June 26 – July 14, 2006	CNA I	PR
October 2, 2006	WANO – Paris Centre	PR Exit Meeting
February 26 – March, 2007	CNE	TSM Methodology of PR
May 23-27, 2007	CNE	PR Introduction meeting
June 19-20, 2007	CNE	TSM Safety Culture
June 21-22, 2007	CNA I	TSM Safety Culture
June 25-29, 2007	CNA I	TSM Operative Experience Management
September 3-21, 2007	CNE	PR
October 16-19, 2007	CNA I	TSM Task Observations
October 22-24, 2007	CNA II	TSM Safety Culture
October 25-26, 2007	NA-SA Head Quarter	TSM Safety Culture
November 6-16, 2007	CNA I	TSM Radiological Protection & Industrial Safety
November 19-23, 2007	CNE	TSM Auto-evaluation
December 18, 2007	CNE	PR Exit Meeting
March 5-7, 2008	CNE	TSM Roth Cause

PERIOD	HOST	ACTIVITY
April 14-18, 2008	CNE	TSM Production Management
April 26 – May 3, 2008	CNE	TSM on Pre-working sessions
June 9-13, 2008	CNE	TSM Materials
August 21-29, 2008	CNA I	TSM on Conduct of Operations
April 13-17, 2009	CNE	TSM on Conduct of Maintenance
April 20-25, 2009	CNA I	TSM on Reduction of Maintenance re-execution
April 27-30, 2009	CNE	PR Pre-visit
July 13-17, 2009	CNA I	PR Follow-up
September 7-10, 2009	CNE	TSM on Teaching Practice for field training
September 14-18, 2009	CNA I	TSM on Chemical cleaning
December 7-11, 2009	CNE	PR Follow-up
December 03 to 21, 2012	CNA I	PR
May 2013	NA-SA Head Quarter	TSM on Corporate Nuclear Oversight
September 2013	CNA II	Pre Start up PR

Other activities carried out between the Licensee and WANO from 2010 to 2013 are listed in article 3.19. Previous activities were included in the National Report 2010 to the CNS.

Other international Missions

- IAEA review mission to the CNAI Spent Fuel Dry Storage Project:
The Agency developed the Irradiated Fuel Management Advisory Programme (IFMAP); the first IFMAP mission was conducted in March 2012 to the CNA I's Spent Fuel Dry Storage Project. An international team of experts reviewed the project's technical and planning documentation, suggested improvements to the conceptual engineering, and issued a report covering technical and organizational issues. The report included recommendations for interaction with the regulator, and technical improvements such as an enhanced drying procedure. Backup options in case of delays, including the temporary use of the spent fuel storage pools of the CNA II NPP adjacent, were also discussed.
- IAEA-Commissioning Procedure Review in CNA II 26-28 October 2011.
- IAEA Workshop on Improved RPV Structural Integrity Assessment for CNA I and CNA II, 18-20 March 2012. Buenos Aires, Argentina.
- CANDU Energy "Severe Accident Management Guidance (SAMG) Training and Validation Exercise for the Severe Accident Management Program", December 2012. Embalse, Argentina.

2.9. PLANT SAFETY UPGRADES

INSTALLATION OF PASSIVE AUTOCATALYTIC H₂ RE-COMBINERS (PARS)

The main safety objective is to maintain the containment integrity during severe accident conditions that originate significant releases of combustible gases by avoiding the hydrogen deflagration /explosion. The CNA I and CNA II containments are in the process of installing PARs provided by AREVA and, for CNE is planned to be installed as part of the plant refurbishment.

For the preliminary design, were selected recombination capabilities and locations of equipment according to AREVA's experience in pre-Konvoi and Konvoi NPPs and taking into account the singularities of the CNA I and CNA II containments verified during a walkdown conducted by AREVA-Gesellschaft Für Anlagen Und Reaktorsicherheit (GRS).

The PARs detailed engineering and its proper functioning verification was carried out by AREVA-GRS. Besides, MELCOR code was used for these purposes.

The above mentioned included the design of support structures location, system description and stress analysis considering dead, thermal and seismic loads.

The PARs number was calculated taking into account a SBO and different LOCAs as initiating events, resulting in 32 PARs for CNA I and 54 PARs for CNA II.

During the 2013 programmed outage, 27 PARs were installed in CNA I, and it is foreseen to install the rest by the end of 2013. In CNA II the foreseen PARs will be installed during 2013.

The PARs foreseen for CNE will be installed in the framework of the plant life extension during 2014.

Strainers' Emergency Cooling Injection

In the light of studies carried out by the international nuclear community about the estimation of the debris load generated for a LOCA, the original strainers installed in the ECI of CNA II were reviewed and it was decided to incorporate filter elements that meet the new international requirements.

Taking into account the CNE's previous experience in this topic, NA-SA contracted to AECL for the design and delivery of the strainers. AECL has an own design strainers known as Finned Strainer which have already been implemented in about 50 PWR NPPs in the USA and France. For the strainers design, NA-SA has considered the following cases:

- A "0.1 A" break in the cold leg coincident with an ECI system train unavailable due to maintenance and another one unavailable by single failure.
- A "2 A" break in the cold leg considering that all ECI system's trains are available.

The strainers' design was performed according to internationally accepted design criteria, considering the breaks which can generate the highest amount of debris. Conservative assumptions were taken into account in order to estimate the total amount of debris that can be transported to the reactor building sump.

The strainers' height is dimensioned such that it can operate with the available water in the sump for the postulated scenarios, and filter surface is calculated and dimensioned with a conservative margin. The strainers in the safety pumps suction line have already been installed.

Upgrade of Emergency Power System at CNA I

The original EPS design includes a set of three emergency DGs with 50% capacity each one of them. Additionally, CNA I has an interconnection line to two of the CNA II DGs that was specially implemented few years ago to improve the CNA I EPS reliability.

Owing to the advanced stage of the CNA II commissioning, considering that the mentioned DGs connected to CNA I have to be reintegrated to CNA II, aimed to maintain the current reliability of the CNA I EPS, it was decided to construct a new EPS system which is housed in a new separate building, equipped with 3 new DGs, with a capacity of 4,000 kVA each one, feeding to the 6.6 kV bus bars. The structure of the building is designed according to the quality standards requirements for the nuclear industry considering earthquake, tornadoes and fire. The DGs are air cooled and their main features are the following:

- Designed with two redundant and diverse trains of 100% each, three autonomous DGs with 100% capacity, one for each redundancy and the third one is in reserve. DGs room physically separated, along with their operating and control systems and having its own fire protection.
- The loads connected to the interruptible "short time" bus bars and the non-interruptible ones will be relocated to sustain the design criteria.
- Two 100% uninterruptible power subsystems and a third unit on standby. This maintains the original design criteria of having a non-interruptible power supply to a set of essential loads, such as valves closing the containment in case of large LOCA and controlled steam release valves.

The modifications were to reduce common cause failures; improved protections; improved redundancy, improved ignition system and improvements in DGs refrigeration. In addition, improvements were achieved in the logic of power (several signals, redundant and within the reactor protection system).

The impact of the new EPS on the CNA I safety was analysed by using both deterministic and probabilistic approaches. The above mentioned analysis consists in comparing the reliability assessment between the new and the original EPS designs as well as in performing a qualitative study of the impact of the new EPS in the scenarios modelled in the PSA. The results of this study indicated that the implementation of the new EPS will improve the plant safety.

The assembly activities are practically finished and the commissioning of the auxiliary systems has already started. It is planned that by the end of 2013 the system will be finished and tested but, the connection to the plant is estimated to be done in 2014 (see section 3.14.3.1.1.1.).

Recently a review of PSA L1 model has started. The system will be incorporated into fault trees and event trees, focusing on electrical headers to quantify its impact in plant safety.

Due to the stress test results and WANO recommendations, a mobile DG (MDG) has been included as a backup to feed safety related loads (pumps, venting valves, etc.) to use in case of severe accident scenarios. The MDG is already in the site, and it is expected the connections to be installed by the end of 2014 (see section 3.14.3.1.1.2.).

COMPLETION OF LEVEL 2 PSA

The Argentinean regulation applicable to the licensing of new NPPs requires that the grouping of accidental sequences must have an annual probability of occurrence which, plotted as a function of the corresponding consequence given by the effective dose, results in a point located in the acceptable zone of the "criterion curve for the public" included in the Regulatory Standard 3.1.3. - Radiological Criteria Relating to Accidents in NPPs. Therefore, a full PSA L2 and a partial PSA L3 are necessities to demonstrate that the probabilistic criterion is fulfilled.

CNA I

The PSA L2 will be required as a condition for the plant life extension project. However, for the SBO sequence, a qualitative comparison of the evolution of the reactor core degradation phenomenology was performed based on CNA II results. The following summarizes the relevant aspects of such study.

- The interval phase that takes place in the RPV is slow, with characteristic longer times. Core degradation is delayed mainly due to the existence of the moderator, which represents a major heat sink. After the relocation of the molten material to the RPV's lower dome, the failure is considerably delayed due to the presence of the filling material, to which a substantial part of the power generated by heat decay is transferred.
- The fuel heating increase rate after its exposure is estimated to be in the order of $0.1 \text{ }^{\circ}\text{K} / \text{s}$, given that the generated power per channel are similar for both plants. Note that the heating increase rate is one of the most important parameters to characterize the RPV in-vessel severe accident progression. The estimated value implies a slow fuel heating, and a longer time for the zircaloy oxidation.
- No core degradation process along with the formation of a molten pool as occurred in the TMI-2 is expected, due to the presence of the cooling channels and to the spacing between them.
- The total hydrogen production prior to the RPV is expected to be around 460 kg, being estimated that about 50% of the zircaloy is oxidized.

CNA II

The PSA level 2 main objective is the determination of the radionuclides release into the environment (source terms) in combination with their annual probability occurrences which are the needed input for the PSA L3.

The PSA L 2 has been produced by the Licensee with the GRS (Gesellschaft für Anlagen-und Reaktorsicherheit, Germany) as TSO, according to the state-of-the-art. The approach corresponds to the IAEA and the German PSA guidelines recommendations.

Deterministic thermo-hydraulic analyses for the PSA L 2 rely largely on computer simulations using the integral code MELCOR one of the most advanced computer codes for severe accident analyses.

The starting points of the PSA level 2 are the accident sequences in the PSA level 1 which reach a core damage state (CDS). The interface between PSA level 1 and level 2 is performed as follows: The CDS sequences are characterized by a list of attributes. All CDS sequences, which have the same set of attribute values, are grouped together. The list of these CDS groups together with the associated frequencies constitutes the input to the PSA level 2. The attributes used are listed in the section 3.14.3.2.2.

The attempt of the PSA level 2 was to prepare a deterministic MELCOR analysis for each CDS. Due to the large number of CDS in the PSA level 1 it was no longer possible to perform a MELCOR run for each of them. It was selected representative transients and LOCAs to perform MELCOR runs. Details are in the section 3.14.3.1.2.7.

Since deterministic analyses as well as the test and experiments results are not sufficient to identify accident sequences and source terms with satisfactory precision and completeness; probabilistic accident progression analysis has to be performed.

The code EVNTRE has been chosen for evaluating the accident progression event tree. The event tree consists of approximately 100 branching points which are binned into nine so-called release categories at the event tree end.

CNE

The Licensee has contracted the services of CANDU Energy Inc. (former AECL), to perform the PSA L2 which was required as a condition for the plant life extension (PLEX) project.

Based on the updated PSA L1 (which includes all improvements for PLEX), the above mentioned PSA L2 is being developed in order to evaluate the containment response and to determine the large early release frequency (LERF). The scope of this work is similar to other CANDU 6, and it includes only internal events at full power.

The methodology for conducting a Level 2 PSA for the CNE is based on internationally recognized practices for CANDU reactors and consistent with the guidance given in the IAEA documents such as SSG-4, 50-P-8 y TECDOC – 1229 for conducting a Level 2 PSA.

The severe accident progression analysis will be performed with the approved version of MAAP4-CANDU code.

The uncertainty analysis will be performed with Risk Spectrum Analysis Tools (RSAT); it is separated software. The uncertainty analyses are performed using Monte Carlo simulation.

REPLACEMENT OF PRESSURE TUBES, FEEDER PIPES AND STEAM GENERATORS DURING THE CNE REFURBISHMENT

During the CNE refurbishment the following activities related with pressure tubes, feeders and steam generators are planned:

- Re-tubing: the process consists, in the removal and replacement of the pressure tubes (PTs) (including the corresponding end fittings and others PT's accessories); calandria tubes (CTs); and feeders, in a planned manner according to the defined logistics. The re-tubing pre-project activities that are being carried out include:
 - o Review of the reactor building and the service Building as well as the surrounding plant structures to determine the capabilities and limitations related to re-tubing logistics.
 - o Specific dimensional and radiation field measurements of the site.
 - o Conceptual design of a permanent waste storage facility and waste characterization.
 - o Qualification of local suppliers of components for the re-tubing process.
 - o The preparation process continues being developed under the technical support of CANDU energy/ former AECL (plant designer) relative to organization, radioactive waste management and equipment and training provision. The lessons learned from similar tasks performed at Point Lepreau and Wolsong I NPPs are being applied.

The PTs are manufactured by a local supplier (CONUAR) which has got the required qualification for manufacturing process of them as well as, the CTs and other reactor internals (see section 3.14.2.2.).

Regarding the feeders, the existing ones will be replaced by new ones with a material different (Chrome, phosphor and sulphur content will be lowered) from that were provided by the original

design, in order to improve structural resistance as well as the resistance to corrosion by minimizing flow-assisted corrosion (FAC). Besides, currently the pre-production samples are under the corresponding certification process (see section 3.14.2.2.3.).

STEAM GENERATORS (SGS) REPLACEMENT:

This task includes the replacement of the cartridges (lower assembly of the SGs), the tube bundles, the pressure vessel envelope, and the steam drum internals (primary moisture separators) for the four SGs.

SGs Replacement was not planned in the original refurbishment program, but it was included because of the extensive degradation detected in the tube support plates affected by FAC phenomena. It was also determined that “the primary moisture separators” need to be replaced since the current ones are not suitable for long term operation. Only the steam drum including some internals will remain from the original SG after replacement is performed.

The new SGs are being manufactured in Argentina under License and supervision of B&W by the local suppliers IMPSA, FAE and CONUAR, which were qualified under the plant designer supervision. (see section 3.14.2.2.4.).

Also, in order to remove and replace SGs in a safe manner, strategies have been evaluated using 3D computational models. It has been decided to use the main airlock to remove and ingress components in order to keep the containment integrity.

The replacement approach involves steam drum cutting and manipulating of old and new cartridges through the main equipment airlock.

The design of the new SGs includes the following aspects: the re-powering of the plant; the site seismic requalification and; the design changes arising from the operating experience related to the PLIM program of the SGs.

2.10. STRATEGIC COMMUNICATION PLAN

COMMUNICATIONS WITH THE STAKEHOLDERS

ARN has the legal obligation to inform the public and the willingness to communicate with stakeholders. In the framework of Decree N° 1.172/03 and bearing in mind the guiding premise of open, straightforward and candid communication, ARN launched a medium-term Strategic Communication Plan (SCP) to optimize the communication channels with society.

Regarding external communication, the SCP's objectives are to foster public understanding of ARN's role and activities; to work on the perception of risk by the public; to develop a sectorized policy of training and information in nuclear matters, 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.

For these purposes, it became clear that the use of Internet and Website based technological approaches fosters a bond of confidence and trust, thus creating a positive image in the public mind with the aim of generating commitment with our public.

ARN regularly issues publications which are aimed at keeping stakeholders informed of the activities undertaken by the institution. In ARN's Website (www.arn.gob.ar), regulatory information, Regulatory Standards, laws and acts, permits and operating license information, Annual Reports, National Nuclear Safety Reports and general public communications among others, can be found.

COMMUNICATIONS IN THE PLANNING ZONE

A revision to the communications on and off-site were performed, among all the government organizations that may be involved, taking into account the lessons from other non-nuclear events.

As result of this revision, the following improvements were proposed:

- The operator of nuclear power plants has initiated under requirement from ARN a study for the construction of local emergency control centres, outside the planning zone of the nuclear emergency (beyond 10 km). These centres will have infrastructure, instrumentation and communication systems in accordance to the lessons learned from the Fukushima accident.

- *In collaboration with other response organizations (mentioned in section 5.1.3.1.6), the operator has reviewed the robustness of the communications outside the site during a severe accident. The result of this review allowed corroborating that the communication system is accordingly robust to deal with such scenarios.*

COMMUNICATION TO ENVIRONMENTAL AUTHORITIES

Within the framework established by Article 9 of the Convention on Nuclear Safety and on the mechanisms by which the licensee maintains an open and transparent communication with the public, 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 foresees the potential citizen participation, non-binding, during the public hearing stage.

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

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

With regard to the first, in May 2012, NA-SA has completed the delivery of the corresponding Environmental Impact Assessment before the authorities of OPDS.

Regarding the CNE Life Extension Project, and in the framework established by the Provincial Environmental Act. N° 7343 and its Decree Regulation N° 2131/00, NA-SA has undertaken with the Ministry of Environment of the Córdoba Province a series of activities (dissemination events, site visits and technical meetings) in order to establish a methodology that facilitates the definition and the agreement of the terms of reference for the corresponding Environmental Impact Study. To date, NA-SA is in the final stage of the service contract for the development of the indicated study.

NA-SA's corporate communications sector carries out a Communication Annual Plan (see section 3.9.3.) whose actions are aimed to maintain an open and transparent dialogue with its stakeholders by means of:

- *Communication campaigns: Mass media campaigns are developed annually.*
- *Relations with the Media: NA-SA maintains strong relationship with the media dedicated to the energy sector, national and local press.*
- *Digital Communication: Active work is being performed in company's digital communication channels.*
- *Relations with the Community: There are annual training programs for teachers and students regarding inherent themes on the application of the Emergency Plan.*
- *Plant Tours An active visit program in the nuclear power plants for external and internal public is permanently on going. In 2012 more than ten thousand visitors were received.*

COMMUNICATION IN THE EMERGENCY CONTROL CENTRE

The communication network from in ARN has the necessary technology to perform teleconferences and/or videoconferences via internet or Integrated Services Digital Network (RDSI). It has a GIS server to transmit information processed in ECC, via internet, to the Local Emergency Operative Centres (COEMs).

A satellite communication system is also available to send information (video, data and voice) between the site of environment monitoring, COEM and ECC. This system includes mobile radiological and meteorological stations, with the possibility to transmit data using a wired network, wireless and satellite communication to dedicated mobile and landline servers to process radiological and meteorological data.

2.11. ENHANCE THE CURRENT STRONG RELATIONSHIPS WITH OTHER ORGANIZATIONS

In the period belonging to the Fifth National Nuclear Safety Report, the relationship between ARN and other organizations remains the same as far as regulatory activities are concerned. In this context, could be highlighted that, the ARN has continued participating mainly in forums and specific Regulatory Bodies' meetings like the Ibero-American Forum of Radiological and Nuclear Regulatory Agencies (FORO), CANDU Seniors Regulators, etc.

In the FORO case, its activity has been particularly intense, and participated, since 2010 until now, in the following meetings:

EVENT	PLACE	DATE
Technical Committee	Montevideo, Uruguay	November, 2010
Technical Committee	Santiago de Chile, Chile	July, 2011
Plenary	Santiago de Chile, Chile	July, 2011
Technical Meeting	Madrid, Spain	September, 2011
Technical Committee	Lima, Republic of Perú	November, 2011
Technical Meeting	Buenos Aires, Argentina	June, 2012
Technical Committee	La Habana, Cuba	July, 2012
Plenary	La Habana, Cuba	July, 2012
Technical Committee	Rio de Janeiro, Brazil	November, 2012
Technical Committee	Madrid, Spain	June, 2013
Plenary	Madrid, Spain	June, 2013

In response to Fukushima accident, the FORO's member countries which having NPPs, being Argentina one of these members, decided to conduct a stress test in each one of their NPPs (see introduction). As a consequence, a Technical Meeting (TM) was held in September 2011 at Madrid, Spain with the main goal of defining the stress test content and scope, the schedule for the preparation of the corresponding National Reports, as well as a cross peer review process. Besides, in this TM was decided to perform a second TM with the main objective of assess the stress tests conducted by the member countries (Argentina, Brazil, Chile, Cuba, Mexico, Peru, Spain, and Uruguay) which was held in Buenos Aires in June 2012. The results of this FORO's TM allowed to conclude that the Argentinean NPPs' have acceptable safety margins against the extreme events considered, not being identified significant weaknesses as to require taking urgent preventive actions including the reactor shut-down.

Argentina's regulatory body participated in the CANDU Senior Regulators Group annual meetings. The last one was devoted to exchange information and lessons learned from the Fukushima Daiichi accident.

The meeting was also useful to exchange information in preparation for the upcoming Second Extraordinary Meeting of the CNS on lessons learned from independent safety reviews (or "stress tests") in response to the Fukushima Daiichi nuclear accident.

With respect to the cooperation with international organizations, it is worth mentioning that after the Fukushima accident, Argentina has participated in many meetings in connection with this incident; as well as it offered support to Japan in different areas.

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

The safety of nuclear power plants (NPPs) in Argentina is continuously assessed and enhanced by acting on results from deterministic and probabilistic safety analysis of operating performance, review of operating experience, audits and inspection results.

The NPPs' Operating Licenses establishes that must be carried out a Periodic Safety Review (PSR) as a necessary condition for the License renewal for a maximum 10 year period, which has to consider the plant safety-relevant aspects. The level of development of the PSR should be equivalent to that indicated in the IAEA document NS-G 2.10 "Periodic Safety Review of Nuclear Power Plants".

3.6.2. EXISTING NUCLEAR POWER PLANTS IN ARGENTINA

Argentina has two NPPs under operating stage, CNA I and CNE, and another under *commissioning*, CNA II. *Besides, the CAREM reactor prototype (CAREM) a low power NPP is under the first commissioning stage.*

CNA I, located some 100 km northwest from Buenos Aires City, with 335 MW (e) began its commercial operation in 1974. The reactor is of PHWR type, with pressure vessel, 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.

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

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

3.6.3. NUCLEAR SECTOR ORGANIZATION

In August 1994 the CNEA was divided into three independent organizations. The first retained the original name, CNEA, it remains within the public sector and its current activities are related to research and development, fuel cycle, radioisotopes and radiation sources, and specialised training in nuclear issues. The second organization is NA-SA (the Licensee, Public Company in charge of the operation of the NPPs), and is constituted by the branch of the former CNEA which was in charge of

NPPs operation and by Empresa Nuclear Argentina de Centrales Eléctricas (ENACE Empresa Nacional de Energía S.A.). NA-SA is also acting as architect - engineer for CNA II.

The third organization, originally named Ente Nacional Regulador Nuclear (National Nuclear Regulatory Board) and later named Autoridad Regulatoria Nuclear (ARN - Nuclear Regulatory Authority) by means of Act Nº 24.804, 1997, is constituted by the former regulatory branch of the CNEA. This branch started its regulatory activities in 1958. The ARN is a completely independent organization, entrusted with all the regulatory functions.

3.6.4. ACTIONS LEADING TO SAFETY IMPROVEMENTS

3.6.4.1. CONTINUOUS EXECUTION ACTIVITIES

There are some continuous execution activities in CNA I and CNE that lead to safety improvements.

Such activities are the same for both plants and comprise:

- Documentation updating.
- Organization updating.
- Components inspection program.
- Periodic Tests Program.
- Emergency Plan (see article 3.16.).
- Training and qualification of operating personnel (see article 3.7.).
- Quality Assurance Program (see article 3.13.).
- Ageing Management Program (see article 3.14.).
- Severe Accident Management Program (see article 3.14.).
- Operating Experience Management Program (see article 3.19.).
- In Service Inspection Program (see article 3.19.).

3.6.4.1.1. Documentation updating

Until 2003 the Argentinean Regulatory Standard AR 3.9.1. –“General Criteria for Operational Safety in NPP” established that the SAR of nuclear installations must be updated each time that a plant design modification be performed, and once every five years. The Operating Licenses, with unlimited renewal period, included similar requirements. In 2003 new license criteria were established with two major changes:

- To include a validity period of the Operating License of no more than 10 years, and
- To require a formal Periodical Safety Review (PSR) for License renewal following IAEA recommendations.

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

In this regard, CNA I is developing a PSR, which implies a review and update of the mandatory documentation, which is being analysed by the Regulatory Body. Also, the Licensee submitted to the Regulatory Body a set of operating instructions to be implemented in abnormal situations related to power supply failures which are being analysed.

In the case of CNE, and as regard of CNE life extension project, the Regulatory Body has additionally required an update of the mandatory documentation, including among others, the Safety Report, the design modifications to be implemented, the plant operating procedures; the preventive and corrective maintenance programs; the repetitive tests program and the inspections program.

3.6.4.1.2. Organization updating

The NPP's organization evolves adapting itself to the successive stages each plant goes through aiming at improving the response to each step's requirements.

In the previous National Report (Fifth) some changes were described regarding to CNA I and CNE as well as to the Embalse Life Extension Project structure. Since 2011 new changes were implemented into NA-SA General Organization Chart (see article 3.13).

There are now the three following Deputy General Managers depending directly from the General Manager:

- *Deputy General Manager of Production –operating;*
- *Technical Deputy General Manager and;*
- *Administrative General Manage.*

The following three Managers depend from the Deputy General Manager of Production:

- *Atucha Site Manager.*
- *Embalse NPP Manager.*
- *Maintenance Support Manager (in charge of technical services to the NPPs).*

The Plant Managers of Atucha I and Atucha II depend directly from the Atucha Site Manager. Besides, the Divisions of Engineering; Radiological Safety and; Training which are common to both plants, also depend from the Atucha Site Manager.

The Divisions of Engineering, Nuclear Safety, Radiological Protection, Environmental Impact, Security, Safeguards, Commercial and Fuel Cycle depend from the Technical Deputy General Manager.

The Departments of Human Resources, Finances, Supplies, and Contracts depend from the Administrative General Manager.

NA-SA has two projects that directly depend from the Board of Directors, namely Atucha II Project and CNE PLEX Project.

The Atucha II Project Director is now also responsible for the Fourth NPP project (currently in charge of analysing different technological alternatives for develop this project).

Organizational Charts for CNA II including construction, commissioning and a common operation organization jointly with CNA I have been proposed.

Finally the contents of an entire Organizational Chart including both CNA I and CNA II was approved by the Regulatory Body but, it is currently being implemented only in CNA I, due to CNA II is in the commissioning stage and as it was mentioned before uses an ad hoc organizational Chart.

3.6.4.1.3. Structures, Systems and Components inspection program

These activities include the follow-up of the reactor components with the purpose of detecting eventual modifications of material properties due to ageing, erosion, fragility, fatigue and defects formation. These modifications may be originated by stress, thermal cycles, temperature, radiation, hydrogen absorption, corrosion, vibrations or friction.

The activities comprise every SSCs of the nuclear and conventional area, which according to applicable standards and operating experience, are considered relevant for the installation safety.

The activities are mainly carried out during both NPPs programmed outages and during shutdown periods lasting several days before start up.

Since 2011, the period of the CNE scheduled outages was modified by a regulatory requirement due to the proximity of its end design life. This requirement consists in shortening the period from 18 to 12 months.

Taking into account the operational experience, some improvements in the In Service Inspection Program were achieved through the acquisition of the COMSY code which is a software tool for the ageing management and plant life management related to mechanical components of NPPs. The analysis of the degradation functions include lifetime prediction regarding to important issues like flow accelerated corrosion (FAC), droplet impingement corrosion, cavitation corrosion, etc. There have been two training meetings about the COMSY code use provided by AREVA: the first one in CNA I in August 2011 and the last one in CNE in February 2012.

3.6.4.1.4. Periodic tests program

Both NPPs have a periodic tests program as part of the surveillance program where the availability of the safety systems and the accident mitigation systems are periodically controlled.

The plan is contained in the Periodic Tests Manual, where procedures, associated documentation, etc. are described. It should be mentioned that the performance of periodic tests in due time and form is mandatory.

The Regulatory Body, through its on-site inspectors, checks the fulfilment of the routine tests program, by witnessing the safety relevant tests, in order to verify that the results meet the corresponding acceptance criteria.

3.6.4.2. SPECIFIC IMPROVEMENT ACTIVITIES

In this section, some safety improvements that were implemented from 2010 to 2012 are shown.

3.6.4.2.1. CNA I improvements

3.6.4.2.1.1. Physical Protection System

A new Physical Protection System (PPS) for Atucha site which includes CNA I and CNA II NPPs is being constructed. This system consists mainly in extend the current CNA I double perimeter fence including CNA II, so that both NPPs are included in a unique protected area.

The PPS includes the installation of latest generation devices. Besides, the PPS will dispose a main control centre and a secondary control centre, as well as a unique access for the personnel and visitors.

Outside the protected area, it is foreseen to implement a zone controlled by a National Security Force, which will be externally bounded by a single perimeter fence having a single access to own vehicles as well as authorized personnel which will enter enabled by security card (proximity card).

3.6.4.2.1.2. New Emergency Power Supply System (EPS)

The original EPS design includes a set of three emergency DGs with 50% capacity each one of them. Additionally, CNA I has an interconnection line to two of the CNA II DGs that was specially implemented few years ago to improve the CNA I EPS reliability.

Owing to the advanced stage of the CNA II commissioning, considering that the mentioned DGs connected to CNA I have to be reintegrated to CNA II, aimed to maintain the current reliability of the CNA I EPS, it was decided to construct a new EPS system which is housed in a new separate building, equipped with 3 new DGs, with a capacity of 4,000 kVA each one, feeding to the 6.6 kV bus bars. The structure of the building is designed according to the quality standards requirements for the nuclear industry considering earthquake, tornadoes and fire. The DGs are air cooled and their main features are the following:

- Designed with two redundant and diverse trains of 100% each, three autonomous DGs with 100% capacity, one for each redundancy and the third one is in reserve. DGs room physically separated, along with their operating and control systems and having its own fire protection.*
- The loads connected to the interruptible "short time" bus bars and the non-interruptible ones will be relocated to sustain the design criteria.*
- Two 100% uninterruptible power subsystems and a third unit on standby. This maintains the original design criteria of having a non-interruptible power supply to a set of essential loads, such as valves closing the containment in case of large LOCA and controlled steam release valves.*

The modifications were to reduce common cause failures; improved in protections; redundancies, ignition systems and in DGs refrigeration. In addition, improvements were achieved in the logic of power (several signals, redundant and within the reactor protection system).

The impact of the new EPS on the CNA I safety was analysed by using both deterministic and probabilistic approaches. The above mentioned analysis consists in comparing the reliability assessment between the new and the original EPS designs as well as in performing a qualitative study of the impact of the new EPS in the scenarios modelled in the PSA.

The results of this study indicated that the implementation of the new EPS will improve the plant safety.

The assembly activities are practically finished and the commissioning of the auxiliary systems has already started. It is planned that in October 2013 the system will be finished and tested but the connection to plant systems will be during the 2014 outage.

Recently a review of PSA L1 model has started. The system will be incorporated into fault trees and event trees, focusing on electrical headers to quantify its impact in plant safety.

Due to the stress test results and WANO recommendations, a mobile Diesel Generator (MDG) has been included as a backup to feed safety related loads (pumps, venting valves, etc.) to use in case of severe accident scenarios. The MDG is already in the site, and it is expected the connections to be installed in 2014.

3.6.4.2.1.3. Other improvement activities

- Probabilistic Safety Assessment (PSA) (details in article 3.14.).*
- Advances in the Severe Accident Management Program (details in article 3.14.).*
- Human Performance Improvements Program: new campaigns for spreading knowledge on Human Error reduction tools, brochures, video transmission of announcements, banners, etc. Almost all the procedures regarding Human Performance were updated during the 2010 to 2012 period, in particular the ones corresponding to: Work Packages; Error Prevention Tools; Tasks Observation and; Self Evaluations.*
- Improvements Program of Self-assessments and Following of Corrective Actions.*
- Since 2012 the Licensee has been implementing quinquennial programs of improvements related with fire protection and emergency preparedness.*
- Replacement of solution of deuteroboric acid for Shutdown System N°2 (SDS # 2) from natural to enriched Boron (B-10).*
- AREVA personnel have performed analysis and field studies to evaluate the installation of a fatigue monitoring system of SSCs.*
- A virtual model of the Main Control Room was developed in order to assess and improve the man machine interface.*

3.6.4.2.1.4. Lessons learned from national and international Operating Experience

In response to national and international safety-significant events and Operating Experience (OPEX), safety assessments are performed by ARN and Licensee staffs. Examples of lesson learned and corrective actions are included in Annex VI.

3.6.4.2.2. CNE improvements

Numerous improvements were implemented during the 2010-2012 period related to the CNE life extension project, as described in section 3.6.5.1.

3.6.4.2.2.1. Physical Protection System

The Regulatory Body, in the life extension project framework, has required improvements to the PPS which are aimed to strengthen the ingress control by installing new gates for vehicles entrance as well as updating the TV cameras surveillance system.

3.6.4.2.2.2. Lessons learned from national and international Operating Experience

In response to national and international safety-significant events and Operating Experience (OPEX), safety assessments are performed by ARN and Licensee staffs. Examples of lesson learned and corrective actions are included in Annex VI.

3.6.5. NPPs LIFE EXTENSION

3.6.5.1. CNE LIFE EXTENSION

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

Therefore, NA-SA has 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 long term operation -LTO- (extended period of 25 – 30 year). The PLEX program is divided into the follow three phases:

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

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

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

The Regulatory Body required to the Licensee, as a condition for licensing the CNE LTO, to carry out a Periodic Safety Review (PSR), following the IAEA guidelines Safety Standards Series No. NS-G-2.1 (see section 3.14.3.3.). Besides, an update of the mandatory documentation including the Final Safety Report was required.

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

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

- **Ageing Assessment.**

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

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

- **Safety Assessment.**

The objective of the Safety Assessment is to determine issues susceptible to improvements, *including among others, updating of the deterministic and probabilistic safety studies, the plant modifications incorporation and the use of updating computational models (see section 3.14.3.). The following are the safety assessments for LTO that currently are being carried out:*

- Periodic Safety Review (PSR): specific requisites were elaborated by ARN according to the IAEA safety guide NS-G-2.10. (see section 3.14.3.3.).

- Review and update of Deterministic Safety Analysis: will be considered new accidental scenarios, updated models and acceptance criteria. Also, reactor repowering, new shutdown trips and others design changes have to be included (see section 3.14.3.1.).
- Probabilistic Safety Assessment: *the PSA L1 actualization was carried out with the objective of incorporating the new plant state after the refurbishment. Therefore, the corresponding design changes to be implemented, as well as the changes arisen from the hypothesis of the plant evolution resulting of the review carried out by the designer. An APS L2 to evaluate the response of the containment and determine the radioactive release frequency is being developed (see section 3.14.3.2.3.).*
- Review of Design Changes implemented in other CANDU plants: as a result of this review several improvements in some issues like trip coverage, emergency power (EPS) and water supply (EWS) systems and ECCS reliability will be implemented during CNE refurbishment.
- Review of CNE design against current national and international standards, regulations and practices.
- Review and update of Hazard Assessments. In particular, a seismic re-evaluation/qualification of the CNE site is included.

The CNE was originally designed for a 0.15 g PGA DBE. Later on (1982/1983) was evaluated that the CNE site seismicity was increased to 0.26 g. Besides, considering the advances in seismic knowledge since the 80's, it was decided to undertake the seismic re-evaluation of CNE.

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

- *Classification of SSCs in three categories according to the extent of the modifications / changes in: Like-Like Replacement, Minor Modification, Major Modification or New Design.*
- *Assigning different seismic qualification requirements for each one of these three categories.*
- *Perform a seismic margin assessment (SMA) to determine potential vulnerabilities of the current design to face seismic event (see section 3.17.2.3.3.2.1.).*
- Severe Accidents. There are two main aspects:
 - Design improvements implementation to deal with severe accident evolutions and conditions. The improvements in the CNE design, related to severe accidents have two sources: the changes recommended by PSA L2 results, and those recommended from the operative experience in others CANDU plants.
 - *To develop and implement Severe Accident Management Guidelines (SAMG) specific for CNE, based on the Generic Guidelines developed by the CANDU Owner's Group (COG), with the designer support. These SAMG will fulfil the Regulatory Body requirements related with the stress test which includes the consideration of the mitigation actions foreseen in order to prevent large radioactive releases as a consequence of damages to the reactor and the spent fuel pool (see 3.14.3.1.2.3.).*
- **Environmental Qualifications (EQ):** A program in order to assure that safety related SSCs will be capable to perform their safety functions during normal and accidental conditions for the LTO is being elaborated by the Licensee. For this purpose, *it acquired the documentation used by the Hydro Quebec enterprise for the analysis and the implementation of the EQ Program in Gentilly II NPP, which has characteristics of design and construction, as well as a similar operation period to CNE. The corresponding procedures are being adapted to the CNE by considering both the degree of applicability and the suitability of the plant existing documents.*
- **Re-tubing:** the process consists, in the removal *and replacement of the pressure tubes -PTs- (including the corresponding end fittings and others pressure tube accessories); calandria tubes (CTs); and feeders,* in a planned manner according to the defined logistics. The re-tubing pre-project activities *that are being carried out* includes:
 - Review of the reactor (RB) and Service Building (SB) and the surrounding plant structures to determine the capabilities and limitations related to re-tubing logistics.
 - Specific dimensional and radiation field measurements of the site.

- Conceptual design of a permanent Waste Storage Facility and waste characterization.
- Qualification of local suppliers of components for the re-tubing process.
- The preparation process continues being developed under the technical support of CANDU energy / former AECL (plant designer) relative to organization, radioactive waste management and equipment and training provision. The lessons learned from similar tasks performed at Point Lepreau and Wolsong I NPPs are being applied.

The PTs are manufactured by a local supplier (CONUAR) which has got the required qualification for manufacturing process of them as well as, the calandria tubes and other reactor internals (see section 3.14.2.2.).

Regarding the feeders, the existing ones will be replaced by new ones with a material different (Chrome, phosphor and sulphur content will be lowered) from that were provided by the original design, in order to improve structural resistance as well as the resistance to corrosion by minimizing flow-assisted corrosion (FAC). Besides, currently the pre-production samples are under the corresponding certification process (see section 3.14.2.2.3.).

- **SGs replacement:** This task includes the replacement of the cartridges (lower assembly of the SGs), *the tube bundles, the pressure vessel envelope*, and the steam drum internals (primary moisture separators) for the four SGs. SGs replacement was not planned in the original refurbishment program, but it was included because of the extensive degradation detected in the tube support plates affected by FAC phenomena. It was also determined that “the primary moisture separators” need to be replaced since the current ones are not suitable for LTO. Only the steam drum including some internals will remain from the original SG after replacement is performed.

The new SGs *are being* manufactured in Argentina under License and supervision of B&W by the local suppliers IMPSA, *FAE* and CONUAR, *which were* qualified under the plant designer supervision. (see section 3.14.2.2.4.).

IMPSA has completed its qualification process and has obtained the ASME qualification to manufacture nuclear components class I. The manufacture of the four new SGs is in progress.

FAE (Special Alloys Manufacturing) was certified by ASME and has already finished making all the SG’s tubes.

CONUAR has obtained the certification to manufacture class I nuclear components and the CANDU designer qualification to manufacture each component of the reactor (end fitting, locking cap, shielding cap, CTs and PTs).

Also, in order to remove and replace SGs in a safe manner, strategies have been evaluated using 3D computational models. It has been decided to use the main airlock to remove and ingress components in order to keep the containment integrity.

The replacement approach involves steam drum cutting and manipulating of old and new cartridges through the main equipment airlock.

The design of the new SGs includes the following aspects: the re-powering of the plant; the site seismic requalification and; the design changes arising from the operating experience related to the PLIM program of the SGs.

- **Digital Control Computers (DCCs) replacement.**

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

Currently, the manufacture of the DCCZ computer was finished and are being performed the corresponding verification tests both to the hardware and to the software behaviour.

The software acceptance tests, by CANDU Energy, are disclosed in a specific document. These tests include the new version of the operating system, changes due to plant repowering, gateway system, new features, etc.

In addition, CANDU Energy will provide the enlargement of the contact scanners for the Main Annunciation System.

- **PLEX activities performed during planned outages.**

The following activities were planned responding to specific objectives of the PLEX program: repair works on piping supports and concrete structures according to the findings of the walk-downs performed during the 2005 planned outage; repair works on the containment structure; walk-downs on Balance of Plant and Nuclear Plant (NP) areas, seismic walk-down, activities related to the logistic for re-tubing SG replacement, SG visual inspections, piping ultrasonic inspections and other inspections recommended by ageing assessment reports.

Besides, in advance to the planned refurbishment outage, as was above mentioned, during last scheduled outages, some recommendations resulting from plant state evaluations have been implemented, and the rest are planned to be done before the refurbishment outage. Basically these tasks have included, among other, to inspections and repairs / reinforcements in pipelines and in containment penetrations, seismic walkdowns, dimensional and radiation field measurements associated with the replacements of the pressure tubes, the SGs and the moderator heat exchangers.

- **Replacement of the Moderator System Heat Exchangers (HXs)**

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

- **Class III DGs replacement**

See section 3.14.3.1.1.1.

- **PLEX activities performed in 2010/12**

The phase 2 is on-going and currently are being manufactured the components to be used in the re-tubing; the construction of the radioactive waste storage facility for the wastes that will be generated during the refurbishment scheduled outage; the preparation of the design packages with the detailed engineering. Besides, recently have been started the personnel training involved in the re-tubing activities.

Moreover, the CNE PLEX project has received the following external assistance:

- *Three experts for site evaluation of CONUAR storehouses (January 2013).*
- *One expert to assist in the process of qualification for upper feeders curved samples (February / March 2013).*
- *One expert to assist in the process of qualification of weld samples and curves for upper and lower feeders (April 2013).*
- *One CANDU expert to assist in the planning of re tubing activities (May / June and July 2013).*
- *Two CANDU experts to assist in feeders samples essays / tests April 17 to June 14 2013.*

3.6.5.2. CNA I LIFE EXTENSION

The Licensee has presented to the Regulatory Body a preliminary report on the feasibility of conducting a PLEX for CNA I which includes mainly a current plant state assessment and the tasks that have been made since its commissioning which resulted in plant safety improvements. Besides, the organization to conduct the PLEX activities as well as the tasks proposed for the different stages were presented. Additionally, the Licensee is analysing an alternative for managing the life extension for subsequent periods of 5 years.

The Regulatory Body is evaluating the above mentioned preliminary report.

3.6.6. OPINION CONCERNING THE OPERATION CONTINUITY OF NUCLEAR INSTALLATIONS

During this reporting period, CNA I and CNE were operating with acceptable safety margins, complying with the Regulatory Standards related to design and operation. The level of defence in depth of both NPPs remains acceptable and 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, CNA I and CNE were operating with acceptable safety margins, complying with the Regulatory Standards related to design and operation, and the level of defence-in-depth of both NPPs remains acceptable. Therefore, the country complies with the obligations imposed in Article 6 of the Convention on Nuclear Safety.

3.7. ARTICLE 7: LEGISLATIVE AND REGULATORY FRAMEWORK

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

3.7.1. NATIONAL LEGISLATIVE FRAMEWORK

The National Commission of Atomic Energy (CNEA) was created in 1950 by Decree N° 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 legal regulations defined CNEA's competence also as the Regulatory Body for nuclear and radiological safety matters, particularly in those aspects 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-Act N° 22.498/56, ratified by Act N° 14.467 and Decree N° 842/58.

Act N° 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.

Furthermore, Decree N° 842/58 approved and puts into force the Regulation for Using the Radioisotopes and Ionizing Radiation and made it effective to govern the use and application of radioactive materials and the radiations they emitted or which were originated by nuclear reactions and transmutations, as well as to sanction in case of violations. 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 Act N° 23.696/89 and by Section 99 Paragraph 1 of the Constitution, created the National Board of Nuclear Regulation (ENREN) by Decree N° 1.540/94, with territorial jurisdiction along the whole national territory, to perform all the regulatory and control functions of the nuclear activity (formerly within the competence of CNEA's regulatory branch), transferring the corresponding staff, equipment and facilities from CNEA to ENREN.

As from 1997, the ENREN adopted the present denomination of Nuclear Regulatory Authority (ARN). In this sense, Act N° 24.804/97 (National Law of Nuclear Activity) was passed, proclaiming the creation of the ARN as Regulatory Body. The ARN is in charge of nuclear activity regulation and control, concerning radiological and nuclear safety, safeguards and physical protection, giving in addition, advice to the National Executive Power on subjects of its competence.

The ARN, as an autarchic entity within the jurisdiction of the National Executive Power, has full legal power to act in the fields of public and private rights, being the successor of the regulatory branch of CNEA and the National Board of Nuclear Regulation.

Argentina has therefore a legal framework approved by the National Congress that regulates all nuclear activity. The administrative and regulatory structure that has been implemented with reference to this issue is constituted in the following manner:

- An Independent Regulatory Body.
- An appropriate set of Regulatory Standards focused on radiological and nuclear safety.
- A system to grant licenses.
- A control system to verify the compliance with the Regulatory Standards and radiological and nuclear safety requirements.
- A sanction system for cases of non-compliance of licenses, standards or other requirements.
- A clear assignation of responsibilities.

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 in the nuclear field for the State.

These are strict commitments and obligations regarding the control of: (a) the non-proliferation of nuclear weapons; (b) nuclear and radiological safety; (c) spent fuel and radioactive waste safe management; (d) physical protection of nuclear materials (e) cooperation in case of nuclear accidents and radiological emergencies and (f) Liability for Nuclear Damage.

In addition, the following laws, decrees and resolutions are a part of the mentioned legal framework:

Act Nº 24.804, enacted in 1997. This Act determines that the National State will establish the nuclear policy and perform research and development activities through the CNEA and regulatory and surveillance actions through the ARN, successor of the ENREN. The law also provides that the CNEA is the national organization which, among other duties, advises the National Executive on the definition of the nuclear policy, is responsible for radioactive waste management, determines the manner in which nuclear power plants and any other relevant facility shall be decommissioned and holds the ownership of the special radioactive fissionable materials contained in irradiated fuel elements. Annex I to Decree Nº 1.390/98 regulates Act Nº 24.804 and establishes that the ARN will be in charge of approving the contingency plans in case of nuclear accidents, programs to face emergencies, and training of workers and public.

Act Nº 25.018, enacted in 1998, determines CNEA's responsibilities as Responsible Organization for Radioactive Waste Management. It also provides that CNEA shall perform the corresponding activities observing the restrictions established by the ARN, complying with the corresponding national, provincial and municipal regulations.

Resolution ARN Nº 9/2003, issued on June 6, 2003, modifies the way of licensing the facilities Class I (relevant) passing from an undefined or permanent term, to an expiry date for the nuclear reactors operating licences. This was the criteria included to grant the operating licenses of Atucha I Nuclear Power Plant and Embalse Nuclear Power Plant.

Decree Nº 981/05, instructs NA-SA, as Licensee, to conform the Atucha II Unit of Management, whose objective will be to carry out the Acts that are required for putting into operation CNA II.

Decree Nº 1.085/06, maintains operational the regime restored for the execution of works on CNA II, granted to the Atucha II Unit of Management.

Decree Nº 1.107/06, declares of national interest the construction and operation of the CAREM reactor for the nucleus-electrical generation of energy.

Decree Nº 217/2006: The Treasury Secretariat of the Ministry of Economy and Production is empowered to grant new loans to finance the completion of Atucha II Project.

Resolution ARN Nº 107/2007 issued on October 9, 2007: Limits the collective doses per unit of practice for nuclear power plants applicable to the discharge of radioactive effluents from nuclear power plants.

Decree Nº 634/2007 approves the system of regulatory rates for New Nuclear Power Plants, which includes the rates for licensing and inspection tasks performed by the Nuclear Regulatory Authority as regards the construction, operation and starting up of the nuclear power plant.

Act Nº 26.566 promulgated on December 17, 2009, declares of national interest the activities for the Embalse Nuclear Power Plant life extension.

3.7.2. NORMATIVE FRAMEWORK

3.7.2.1. INTRODUCTION

Act N° 24.804/97 empowers the Regulatory Body to issue and establish the standards, which regulate and control nuclear activities, of compulsory application, along the whole national territory.

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

In the course of time a normative system was established comprising subjects such as radiological and nuclear safety, safeguards of nuclear materials and physical protection. The system, known as "AR Standards" (AR Standards for Regulatory Authority), has at present 64 regulatory standards of which 30 concern directly or indirectly to NPPs in all their stages: design, construction, commissioning, operation and decommissioning. The codes and names of the before mentioned 30 regulatory standards are shown in Table 3.7.1.

Additionally, there is a permanent Regulatory Body activity that constantly reviews and updates the regulatory standards with the objective of maintaining the normative system updated.

Table 3.7.1. - AR Standards concerning nuclear power plant licensing:

AR Code	Name
0.0.1.	Licensing of Type I Installations
0.11.1.	Licensing of personnel of Type I Installations
0.11.2.	Psychophysical aptitude requirements for Specific Authorizations
0.11.3.	Retraining of personnel of Type I Installations
3.1.1.	Occupational exposure in nuclear power plants
3.1.2.	Limitation of radioactive effluents in nuclear power plants
3.1.3.	Radiological criteria relating to accidents in nuclear power plants
3.2.1.	General safety criteria in the design of nuclear power plants
3.2.3.	Nuclear power plant fire protection
3.3.1.	Nuclear power plant reactor core design
3.3.2.	Nuclear power plant heat removal systems
3.3.3.	Nuclear power plant primary pressure circuit
3.3.4.	Nuclear power plant fuel performance
3.4.1.	Safety-related protection and instrumentation system in nuclear power plants
3.4.2.	Nuclear power plant shutdown systems
3.4.3.	Nuclear power plant confinement systems
3.5.1.	Emergency electric power supply in nuclear power plants
3.6.1.	Nuclear power plant quality system
3.7.1.	Documentation to be submitted to the Regulatory Authority prior to the commissioning of a nuclear power plant
3.8.1.	Pre-nuclear commissioning of nuclear power plants
3.8.2.	Nuclear commissioning of nuclear power plants
3.9.1.	General criteria for operational safety in nuclear power plants
3.9.2.	Communication of significant events in nuclear power plants
3.10.1.	Protection against earthquakes in nuclear power plants
3.17.1.	Nuclear Power Plant decommissioning
10.1.1.	Basic Radiation Safety Standard
10.12.1.	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 eight regulatory guides; four of them are related directly or indirectly to NPPs. The list of the mentioned guides is shown in Table 3.7.2.

Table 3.7.2. - AR Regulatory Guides concerning nuclear power plants

AR Code	Name
AR 1	Dosimetric factors for external irradiation and internal contamination and emergency levels in food
AR 3	Specific functional conditions to be verified by the specialized physician according to psychophysics performance score
AR 8	<i>Clearance generic levels</i>
AR 10	Training programme of personnel for Type I Installation

Remarks: Table 3.7.2. corresponding to the National Nuclear Safety Report 2010 has been modified as follows:

- The capital letters in the code of ARN Safety Guides have changed from GR to AR.*
- The GR2 Safety Guide is not valid since the end of 2012 because it contained recommendations on how to comply with the safety requirements of a superseded version of the Regulatory Standard AR 10.16.1. "Transport of radioactive materials". The updated version of the Regulatory Standard AR 10.16.1. has adopted the 2009 Edition of the IAEA Regulations for the Safe Transport of Radioactive Material.*

The Regulatory Body has continued with the revision of its standards related to NPPs verifying that the corresponding requirements are adequate to be applied to the activities related with the CNE NPP life extension and the licensing process of CNA II and CAREM as well as the future licensing process of the Fourth NPP.

The Regulatory Standard AR 3.1.3. "Radiological Criteria Related to Accidents in Nuclear Power Reactors", which contains the Argentine acceptance criterion - from the safety point of view - applicable to the design of NPPs, was one of the main reviewed standards. As a result, it was decided to develop a guideline for the implementation of that regulatory standard. Meanwhile, a preliminary version of the corresponding guideline is being applied in CNA II licensing process, as a previous and necessary step in the development of the corresponding Regulatory Guide.

The preliminary version of the new Regulatory Standard AR 10.10.1. "Siting of Class I Nuclear Installations" and the corresponding two Guides are being developed and will take account the lessons learned from the Fukushima accident related with NPPs sites when those lessons be incorporated in the IAEA standards.

3.7.2.2. HARMONIZATION WITH IAEA STANDARDS

Argentinean Nuclear Safety Report 2010 indicated that the harmonization process of ARN regulatory standards against IAEA's standards concluded that "Argentine regulatory standards are completely consistent with IAEA's corresponding standards, taking into account that ARN has mainly adopted a performance criterion".

As ARN regulatory activities in Argentina profit from IAEA safety standards, which are a source of permanent consultancy particularly in the regulation of NPPs in Argentina, a second harmonization stage to perform a comparative analysis of the recommendations contained in ARN's Safety Guides is not considered yet an essential task.

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. ARN regulatory standards and guides will take account of those lessons when they are incorporated in IAEA standards.

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.

The risk information data sources used by ARN derive from reliability studies, Probabilistic Safety Analysis (PSA) and risk-informed operational insights.

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. The deterministic approach considers that an installation is safe when its design, construction and operation are able to face any of the events of a set of postulated accidental events, assuming the impossibility of occurrence of unforeseen accidents. Whereas the probabilistic approach considers that any type of accident may occur with a certain probability, including the occurrence of an unforeseen accidental situation.

Both approaches are complementary and the modern trend in radiological and nuclear safety is to use them both together in a balanced manner. In this sense the ARN adopted, more than three decades ago, the probabilistic criterion of risk acceptance while maintaining some deterministic requirements.

Regulatory Standards are not prescriptive but of compliance with safety objectives (performance). How such objectives are achieved is based on good engineering judgement, in the operators' qualification and in the Licensee's way of taking appropriate decisions.

The role of the latter is to be sceptical and critical, without proposing "how", which implies an interaction of intense, continuous and personal nature, among professionals of the Regulatory Body and the Licensee, all along the installation's lifetime, without affecting either institutions independence.

Thus it may be asserted that the effectiveness of the regulatory system adopted by Argentina is reasonable according to the results it has produced in the course of time.

Regarding the adoption of a performance conception, some of the most important advantages, learnt by the verified application experience are:

- The nature of the interaction between the Regulatory Body and the Licensee contributes to the minimization of the possibilities of non-compliance or deficient compliance with regulatory requirements, as well as enables time and effort saving in fulfilling such requirements.
- The solutions to regulatory requirements come, in general, from the Licensee, that is to say, from the organization that knows in detail everything concerning structures, components, equipments and systems constituting the installation involved.
- The establishment of safety objectives helps to face safety questions with a high degree of flexibility rather than the typical prescriptive rigidity, without losing the bulk regulatory strictness. In this way, situations that would not be adequately considered in a prescriptive regulatory system can still be satisfactorily solved in time and form.
- Each non-objected solution indicates a high degree of consensus between the Regulatory Body and the Licensee, so that its implementation becomes extremely easier. It may be mentioned, however, that in some opportunities it is necessary for the Regulatory Body to have some prescriptive criteria in order to verify the performance fulfilment in the solutions proposed by the Licensee.

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. The mere compliance with the Regulatory Standards does not exempt the Licensee from the mentioned responsibility.

The Regulatory Standards establish that the construction, commissioning, operation or decommissioning of a NPP shall not be initiated without the corresponding License, previously required by the Licensee and issued by the Regulatory Body. The validity of such Licenses is subordinated to the compliance with the conditions stipulated in the corresponding License, as well as with the regulatory standards and requirements issued by the Regulatory Body. 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.

The NPP personnel must be adequately trained and qualified according to their functions in the installation.

The Regulatory Body also requires that the personnel assigned to safety related tasks must be licensed.

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.). In particular, the NPP's design must comply with the radiological criteria related to accidents (Regulatory Standard AR 3.1.3.).

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 an iterative 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 fuel and moderator loading, 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 constituted by senior specialists, who continuously evaluate the execution of the commissioning program and recommends its continuation (Regulatory Standards AR 3.7.1., AR 3.8.1. and AR 3.8.2.).

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 analyzing 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 should imply the application of sanctions that could lead to the Operating License 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 NPP's 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, provisions 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 envisages the following permits: Authorization for Use of Site and

Construction, Authorization for Fueling, Authorization for Core Subcritical Testing, Authorization for initial criticality, Authorization for Zero Power Tests, Authorization for Power Increase and Authorization for Full Power Tests. The conditions and requirements for granting such authorizations were defined in a document approved by the ARN Board of Directors. The main difference with a conventional licensing process is given by the possibility to complete the plant design detailed information as it moves forward with its development.

3.7.3.2.2. Periodic Safety Review and License Renewals

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 safety reviews, which are part of the continuous improvement program, foresee a continuous follow-up of the safety problems, 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.

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 necessities for the effective performance of the plant staff duties which includes 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 persons who apply 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, which will depend 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's 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's 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.

3.7.3.3. REGULATORY INSPECTIONS AND AUDITS

From the beginning of nuclear activity in the country, the Regulatory Body has performed 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 N° 24.804, authorizes the ARN to continue with such inspections and regulatory assessments, performed by its personnel such as:

- **Routine 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** 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. They have several purposes, e.g. to control preventive maintenance tasks during a NPP's programmed shutdown.

- 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 programmed 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 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 certain time period.

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

Article 16 of Act N° 24.804 authorizes the ARN to impose sanctions to the Responsible Organization / Licensee in case of non-compliance with the Mandatory Documentation; it means Regulatory Standards, Licenses (Construction, Commissioning, Operating and Decommissioning) and Regulatory Requirements.

The above-mentioned sanctions are set out in the "Sanctions Regime for Nuclear Power Plants" (Resolution N° 63-05/05/99) and are graded according to the importance of the fault and the potential damage as follows: warning, fines, suspension of the license, permit or authorization or its cancellation. For these purposes, the ARN is authorized to lay down the relevant procedures that may apply in case of violation of the Mandatory Documentation to be issued in the exercise of its competence, ensuring the constitutional guarantees of due process and the defence of rights.

The sanction 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, an informal ARN function is to make Responsible Organizations and Primary Responsible aware of their responsibility regarding safety, in order to increase the communication of safety culture through all levels of the organization structure.

3.7.5. COMPLIANCE WITH THE OBLIGATIONS IMPOSED BY THE CONVENTION

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

- An appropriate set of Regulatory Standards to be applied in safety subjects.
- A licensing system.
- An assessment and inspection system to verify compliance with the mandatory documentation (Licenses, Standards, Requirements, Licensee documentation as SAR, Policies and Principles manual, etc.).
- A sanction regime to be applied in case of non-compliance with mandatory documentation.

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

3.8. ARTICLE 8: REGULATORY BODY

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

3.8.1. FUNCTIONS AND COMPETENCE OF THE REGULATORY BODY

Since the initial operation of Argentina's first research reactor in 1958, a sustained nuclear research and development (R&D) has been carried out in the country, which required the qualification of specialists in several subjects. During the first years, this aim was accomplished by training the professionals abroad, but the country was soon able to satisfy its main needs. CNEA had already reached a reasonable degree of R&D in the nuclear field and a suitable technical-scientific capability to face the R&D of each of the nuclear fuel cycle stages, including the corresponding radiological and nuclear safety, safeguards and physical protection aspects.

Since its creation, the ARN is applying the same policy, i.e., to have qualified and trained human resources to carry out every task implying a regulatory action. Moreover, it has the necessary equipment and laboratories to perform the environmental or biological evaluations enabling a quantitative estimate of both situations and radiological safety of the involved installations.

Act Nº 24.804, sets that the ARN is in charge of the regulation and surveillance of nuclear activities concerning radiological and nuclear safety, physical protection and safeguards. It also establishes that the ARN has autarchy and complete legal capability to act in the field of private and public rights, and that its resources are basically funded through regulatory fees and the State support.

Article 16 of the mentioned Act establishes the functions and responsibilities of the ARN, authorizing it to issue Regulatory Standards concerning radiological and nuclear safety, physical protection and safeguards. This article also assigns to ARN a series of functions, referred to radioactive and nuclear installations, already mentioned in Section 3.7.1. of this report.

Act Nº 24.804 also gives the necessary legal competence to ARN to establish, develop and apply Regulatory Standards to every nuclear activity carried out in the country. In that sense, the Law provides that the regulation and surveillance of the nuclear activity on such matters is "subject to national jurisdiction". The ARN is also responsible for advising the Executive Power on issues under its purview. Furthermore, Article 16 establishes a consulting procedure with the Licensees and the stakeholders every time new Regulatory Standards are proposed or already existing ones are to be modified.

In order to guarantee a proper control level, such legal competence is complemented with a suitable technical capability. Namely, ARN has the capacity to independently evaluate the design, construction, commissioning, operation and decommissioning of NPPs. Therefore, ARN have qualified personnel with adequate knowledge and experience in order to preserve ARN's own independent criterion, regarding every aspect of radiological and nuclear safety.

Furthermore, ARN is empowered to contract specialists who can advise about subjects related to the specific fulfilment of its functions.

Therefore, the global strategy of the regulatory system is concentrated on the following basic aspects:

- Issue of the corresponding standards.
- Execution of regulatory inspections and audits to verify the compliance with granted licenses and authorizations.
- Independent execution of reviews and assessments for the licensing process of nuclear installations.
- Development of technical and scientific knowledge associated to radiological and nuclear safety.
- Training of personnel involved in radiological and nuclear safety, either belonging to the ARN or those working at installations, which perform practices under regulatory control.

It should also be mentioned that ARN has a similar approach in the fields of safeguards and physical protection.

Act N° 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.

It must be pointed out that part of the functions that the Law attributes to the ARN within the Waste Management Area were “de facto” already being executed as a part of ARN’s action plan.

In November 1998, the Executive Power subscribed Decree N° 1.390 defining the scope and procedures facilitating the enforcement of the Act N° 24.804, or “National Law of Nuclear Activities”.

The application of Decree N° 1.390 caused a change in ARN's functions, powers and duties related to radiological offsite aspects concerning nuclear emergencies preparedness, training and response. This change in ARN's functions is explained in article 3.16.

3.8.2. ARN ORGANIZATIONAL STRUCTURE

According to the provisions in Act N° 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*.

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

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

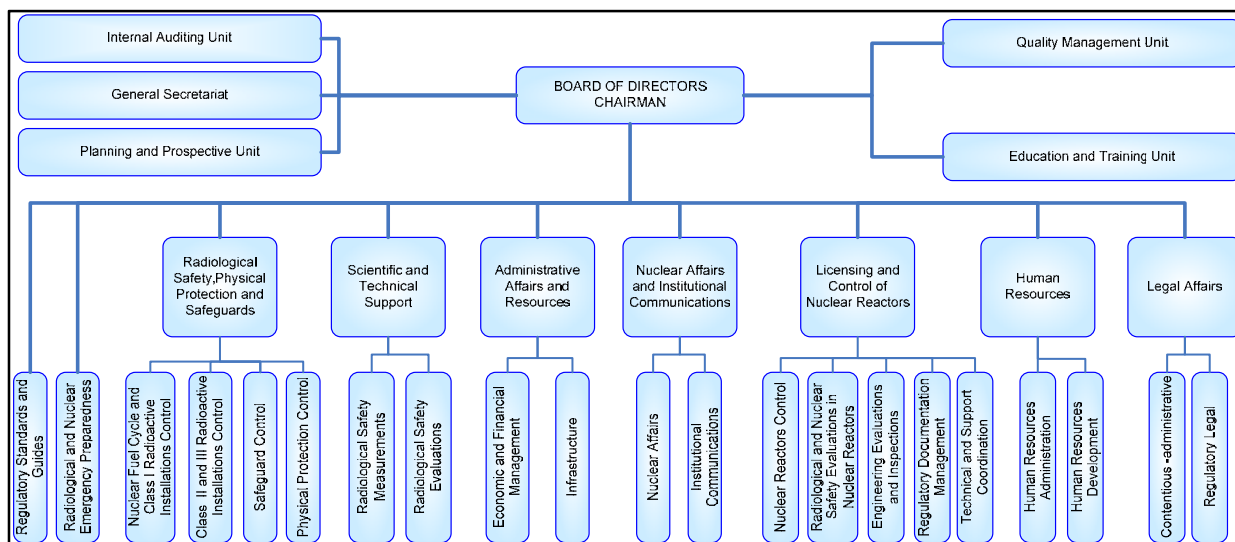


Figure 3.8.1. – ARN Organization Chart

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

The General Secretariat has the main function of assist the Board of Directors on the management tasks and to coordinate the interactions between different areas.

The Units of Internal Auditing and Planning and Prospective advise the Board of Directors on management control results and about the use of financial resources of each project and activity.

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

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

The Radiological Safety, Nuclear Security and Safeguards Division carry out regulatory inspections and assessments concerning radiological safety of radioactive installations (medical, research and industrial installations), transport *of radioactive and nuclear materials*, safeguards control, and *physical protection* controls. *Besides, this division controls the use of nuclear materials, equipment and installations of nuclear interest and verify the compliance with international agreements related to non-proliferation guarantees, as well as* the compliance of physical protection regulations applicable to nuclear materials and installations.

The Scientific and Technical Support Division gives specialised technical support to regulatory inspections and evaluations and carries out *measurements, calibrations and* technical scientific developments on subjects related to radiological and nuclear safety.

The Administrative Affairs and Resources Division provide administrative and accounting support to ARN's regulatory tasks.

The Nuclear Affairs and Institutional Communication Division 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 *division* co-ordinates the institutional relations in the national and international sphere.

The Departments of Reactors Control and Special Processes were unified in the Division of Licensing and Control of Nuclear Reactors. Currently this Division is in charge of the control and licensing of NPPs (CNA I, CNA II, CNE, CAREM and Fourth NPP –see section 3.8.3.1.3.–) and research reactors. Its activities include the design, construction, commissioning and operation stages, as well as the corresponding inspections, regulatory audits and technical assessments.

The Human Resources Division is responsible for the development and administration of the ARN human resources through two Departments: the Human Resource Management and the Human Resources Development.

The Legal Affairs Division participates in the legal aspects of regulatory management, such as licensing, penalties and purchasing processes.

The Regulatory Standards and Guides and the Radiological and Nuclear Emergency Preparedness Departments, reports directly to the Board of Directors.

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

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

3.8.3. ARN HUMAN RESOURCES

During the period covered by this report, the ARN has increased its personnel, from 364 in April 2010 to 400 in March of 2013.

During 2012, it was continued managing the ingress of 30 persons to the ARN's permanent plant, which has been initiated in 2011. At the same time, the management to cover 7 new permanent staff vacancies were initiated. Besides, it was managed through interviews carried out with the Head of the Ministers Cabinet to assess the needs of the ARN related with the staffing corresponding to each organizational unit, aimed to obtain 80 new permanent staff vacancies.

The increase in personnel during the *2010 / 2013* period, mainly young professionals, is due mainly to the need to cover positions which have to be filled as specialised professionals reach retirement age and also to cover the new tasks to be undertaken for the licensing of new nuclear projects in Argentina.

It is worth mentioning that of the *400* people working at the ARN, *52%* are professionals and *18%* are technicians that undertake specialised technical tasks in areas of their competence or are under training programs, and *30%* perform *support* activities. Moreover, 7% of the total ARN personnel hold

high level positions, and have a specialised training of *more than* 20 years or working in regulatory activities.

In relation with the ARN professional staff, the 41% belongs to the engineering branch, 28% to the Natural Sciences area (physics, chemistry, biochemistry, medicine, etc.), 10% is specialized in social sciences, 7% belongs to the economic sciences branch, 4% to legal area and the remaining 9% belong to other specialities.

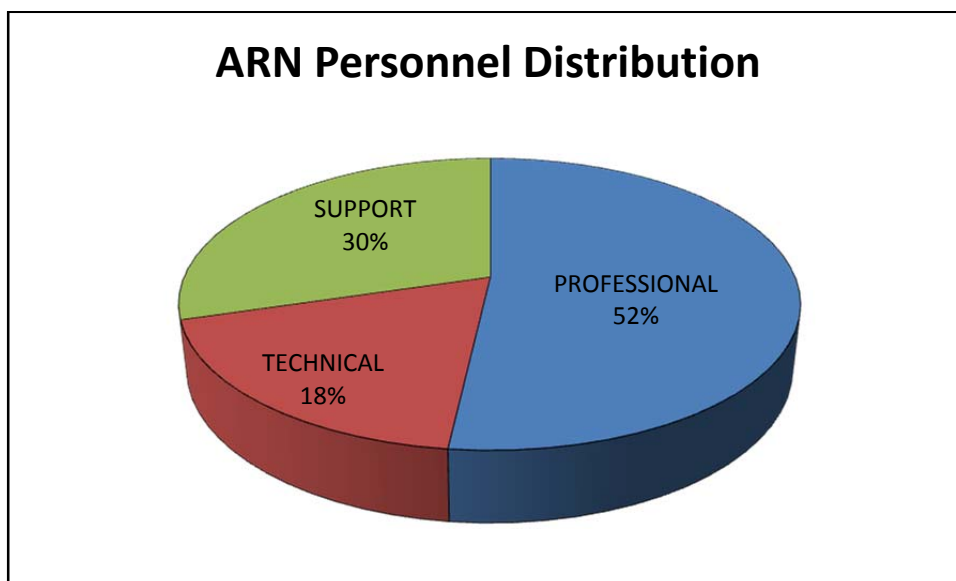


Figure 3.8.2. – ARN personnel distribution

The ARN has the policy of retain specialized technical staffs which has been retired, which continue as advisors in different relevant regulatory issues. Presently, the ARN disposes of 15 advisers.

The whole staff is geographically distributed as follows: *73%* at the headquarters in Buenos Aires City, *19%* in the Ezeiza Atomic Centre, *7%* at CNA I, CNA II and CNE sites, and *1% in other dependencies, which includes* the Argentine-Brazilian Agency for Accounting and Control of Nuclear Material (ABACC) with headquarters in Rio de Janeiro, Brazil and IAEA, in Vienna.

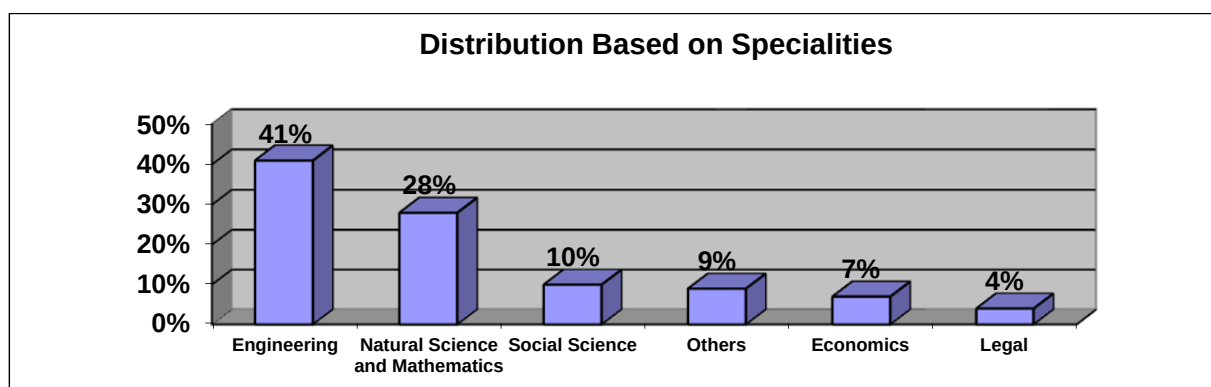


Figure 3.8.3. – Distribution based on specialities

The staff ingress to the ARN permanent plant during the 2010 / 2013 period was a relevant activity which has involved to all the related areas jointly with the Human Resources Division to achieve the main objective which consists in the necessary work stability to the ARN personnel.

In the period reported, the activities aimed to develop the competency profiles required for the personnel has been continued, all the staff was trained on the implementation of competency management and, the Human Resources Development Department personnel has been trained to acquire the necessary skills to monitor the mentioned competency management. The competency management development will allow define the job profiles and to develop a training plan for the medium and long term, according to each post profile.

3.8.3.1. RESOURCES ASSIGNED TO THE NUCLEAR POWER PLANTS REGULATORY CONTROL

3.8.3.1.1. General aspects

In the past, ARN experienced a period with a marked decrease of specialised personnel, but *starting from 2007 this situation has been improved, continuing during the 2010-2013 period, in which significant* resignation of senior personnel hasn't occurred.

As were mentioned in the previous section, in the case of young professionals, has continued with its personnel policy consisting in prioritizing their incoming as well as the corresponding training activities.

In Argentina, the nuclear energy renaissance produced a positive impact in personnel motivation that helps improving the situation, by offering the professionals the opportunity of participation in attractive technological projects.

ARN, in order to comply with its responsibilities, undertakes as a permanent activity training of specialists in radiation and nuclear safety, safeguards and physical protection, by means of training courses and the participation in local and international training activities. These activities are carried out through the *UCE* in charge of defining, organizing and coordinating courses, workshops and follow-up seminars.

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

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

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

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

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

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

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

As can be seen in section 3.8.6., ARN has several agreements with local and international support institutions. One of the conditions of the agreements is that the institutions involved do not provide assistance to the licensees in the same area.

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

Among these institutions, it is possible to mention TÜV (for inspections) and GRS (for evaluations) from Germany and Sandia National Lab (PSAs L2 and L3 assessments) from USA, as well as an agreement with the US - NRC on the use of specific codes. On the other hand, the Argentinean enterprise INVAP (regulatory branch) and the Litoral University are regular TSO on safety assessment issues.

ARN trains its own personnel as well as technical personnel from other national and international institutions on radiological protection and safety of radioactive and nuclear sources (*see section 3.8.3.1.4.*).

Therefore, ARN has the necessary resources and infrastructure to carry out its mission and to reach its objectives.

3.8.3.1.2. Human Resources assigned to operation control

According to the new organization that was mentioned in section 3.8.2., the Control and Licensing Nuclear Reactors Division *is in charge of the NPPs control and licensing* (see Figure 3.8.5.).

About 65 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 2013 is the following:

Inspections and assessments in NPPs	25%
Support activities directly related to safety	40%
Support activities indirectly related to safety	35%

These tasks include two resident inspectors in each NPP, and the safety analysers, who perform supporting tasks to inspection activities and study particular issues regarding the installation safety. *In addition, ARN elaborates an annual Regulatory Audits Plan that involves areas of regulatory interest of CNA I, CNA II and CNE, to verify compliance with regulatory standards, especially the regulatory standard AR 3.6.1. (Quality System in Nuclear Power Plants), as well as all other applicable mandatory documentation compliance.*

3.8.3.1.3. Human Resources assigned to construction and commissioning control

The human resources of the Regulatory Body involved during CNA I's (1969-1974) construction and commissioning stages are different to 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 (ARN since 1994) different experience in such cases.

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:

- preliminary tests.
- material reception.
- tests of components.
- equipment and safety systems fabrication and functioning tests.

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:

- analyse the Safety Analysis Report (SAR).
- analyse the commissioning program.
- analyse the quality assurance program.
- perform and require the performance of the SAR update.
- carry out inspections, audits, assessments and issue requirements to the Licensee.

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 is licensing a second-generation NPP whose Construction Licence was granted in July 1981 and whose construction has stretched over nearly three decades and *has finalized*. During this period, 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 are complied with (see more details in article 3.18.). In 2006 ARN had formed an ad hoc organization for the commissioning of CNA II, in force until December 2009, whose structure is shown in Figure 3.8.4.

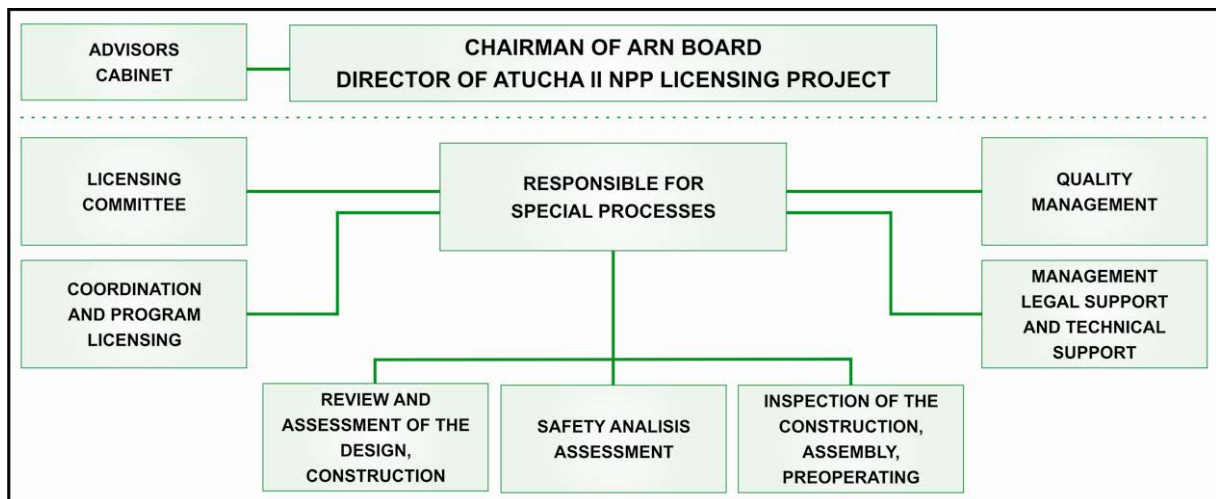


Figure 3.8.4. – Structure of the organization in charge of the commissioning of CNA II (2006-2009)

As it was mentioned in section 3.8.2., the functions of the Special Processes Department were included in the Control and Licensing of Nuclear Reactors Division. The structure of the Control and Licensing of Nuclear Reactors Division is shown in Figure 3.8.5.

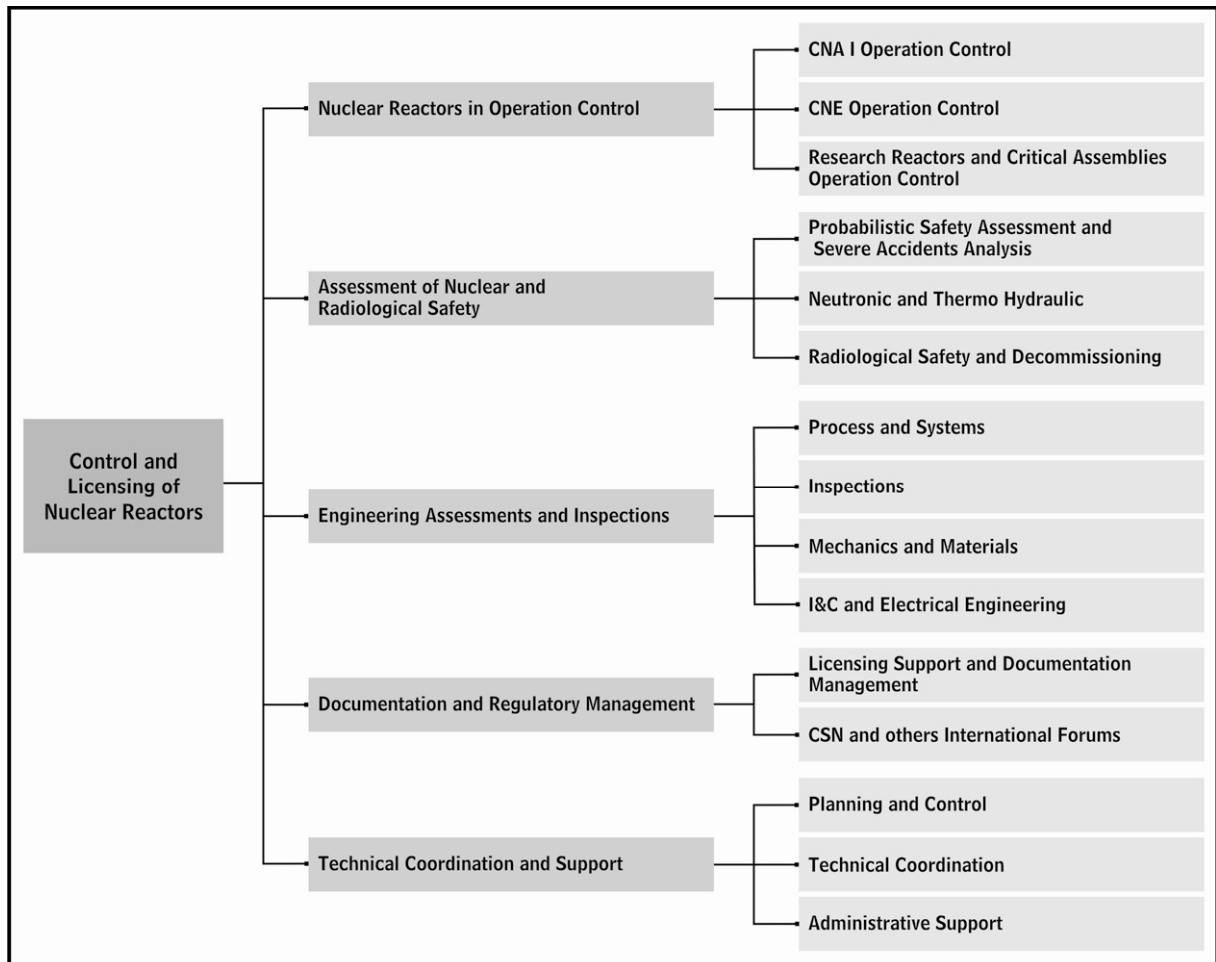


Figure 3.8.5. – Structure of the Control and Licensing of Nuclear Reactors Division

3.8.3.1.4. ARN personnel qualification

The ARN is recognized by the IAEA as the organization in charge of the Regional Training Centre (RTC) in Latin America and The Caribbean for Nuclear, Radiation, Transport and Waste Safety (in accordance with the relevant IAEA's General Conference Resolutions) and based on more than 30 years of successfully conducting Postgraduate Educational Courses (PGEC) on Radiation Protection and Safety. After a successful completion of an EduTA mission, a long-term Agreement between IAEA and the Government of Argentina to support ARN as RTC has been signed in 2008 and is being implemented routinely. During 2010 the recertification of both PGEC by ISO (9001) 2008 has been obtained with regard to their curricula design, planning and development.

In 2010, in order to effectively carry out the important activity entrusted, the Board of Directors of ARN created the UCE to perform the following activities:

- Fulfil the ARN's obligations and develop the corresponding activities as IAEA RTC headquarters, according were foreseen in the mentioned long-term agreement.*
- Propose to the Board of Directors strategies and action courses for the education and training of the ARN staff on topics related to the ARN specific functions as well as to manage the resulting activities.*
- Ensure the institutional knowledge management.*

The activities of the UCE have been focused to its objective of maintaining and enhancing the quality of their PGEC, where the ARN has formed more than 1,000 professionals, of which 45% were Argentines.

The PGEC on Radiation Protection and Safety of Radiation Sources, after an academic scrutiny process, in 2012 was recognized for University of Buenos Aires as a specialization career. Besides, it is expected to fulfil the same recognizing process with the PGEC on Nuclear Safety by the end of 2013.

Other activities managed by the UCE, during the 2010 / 2012 period, related to the formation of regulatory staff were:

- The development of 5 courses of radiological protection for technical level personnel (1 course in 2010 and 2 courses in 2011 and 2012) with 240 hours duration each. In these courses a total of 131 participants have participated, of which 29 participants (22%) are part of the ARN technical staff.*
- The organization of specific internal courses that have been demanded by different ARN's Divisions and Departments.*
- The arrangements to continue with the seminar cycles imparted by expert staff which are near the retirement.*
- The analysis of the external refinement programs applied to senior personnel.*

As were reported in the previous national report, the professional staffs entering the regulatory branches take these PGEC in the early years of its incorporation so that, with the new status acquired, within the first three years would reach a specialist beginner level. Moreover, the majority of senior professional staff having a "practitioner" status would be equated to the quality of a college degree specialist

On a case by case basis, intensive on the job training, specialized short-medium term training courses (domestic and abroad), workshops, fellowships, knowledge management activities and the assignment of tasks under supervision by senior and specialized staff following a graded approach with respect to the complexity of the duties are also performed. Collaborative activities through contracts or cooperation agreements with domestic and foreign organizations to train and cover ARN's new challenges were also performed. Subsequently, those professionals assigned to NPPs inspections and other licensing and regulatory activities receive a wide and complete specific training in NPPs.

ARN's and Licensee's experts take part in safety related activities in organizations like ICRP, UNSCEAR and IAEA (e.g. CSS, INSAG, IRS, NSGC, NUSSC, RASSC, TRASSC, WASSC consultant and technical meetings). They also participate in OSART, IRRS and in international expert missions. This allows a fruitful exchange of experience and lessons learnt and promote a high level of competence in safety.

3.8.4. QUALITY MANAGEMENT SYSTEM IN THE REGULATORY BODY

ARN has implemented a Quality Management System (QMS) according to the requirements established in the ISO 9001 Standard. The actions and requirements fulfilment are described in the Quality Manual (MC-ARN). In this document the ARN Board of Directors declares and communicates the Institution's Quality Policy and Commitment.

Quality Policy

ARN has the commitment of protecting people, environment and future generations from the harmful effects of ionizing radiations and of ensuring the pacific nature of the nuclear energy uses, on the basis of continual improvement of the quality management system effectiveness, process and resource management, personnel excellence with the highest professional training and safety culture development, satisfying the stakeholder requirements.

3.8.4.1. PROCESS APPROACH IMPROVEMENT

The QMS is implemented by applying a process approach.

In 2007 the Board of Directors initiated the discussion, interpretation and implementation of the IAEA Safety Requirements GS-R-3 "Management System for Facilities and Activities".

All ARN processes were evaluated and reviewed regarding the *Management System* requirements in an integrated approach. *As a consequence, the Process Map was developed and approved in 2008 and included in the Quality Manual in 2010 (See Figure 3.8.6).*



Figure 3.8.6. – ARN Process Map

Each process is described in a Process Record where the objectives, inputs, outputs, checkpoints, associated documents, performance indicators among others, are considered.

The measurements and analysis of these processes are carried out by independent assessments and self-assessments. Effectiveness and efficiency indicators are implemented for the certified processes and are being defined for the other processes.

The certified processes, up to May 2013, according to ISO 9001:2008, are the following:

- 1) Curriculum development, planning and performing the postgraduate courses: "Radiation Protection and Safety of Radiation Sources" and "Nuclear Safety" / other training courses and activities on radiation protection, nuclear safety, safeguards and physical protection.
- 2) Protection against ionizing radiation in the transport of radioactive material (TMR Process).
 - Licensing of packages, special radioactive materials and shipments of radioactive material.
 - Development and upgrading of regulatory standards and guides related to the transport of radioactive material.
 - Control of shipments of radioactive material.
 - Advice, training and communication in the transport of radioactive material.
- 3) Licensing staff from Class I facilities and Class II and III nuclear fuel cycle facilities.
- 4) Management of incoming and outgoing documentation.
- 5) Development and revision of Regulatory Standards and Guides.
- 6) Radiological and Nuclear Emergency Preparedness System.

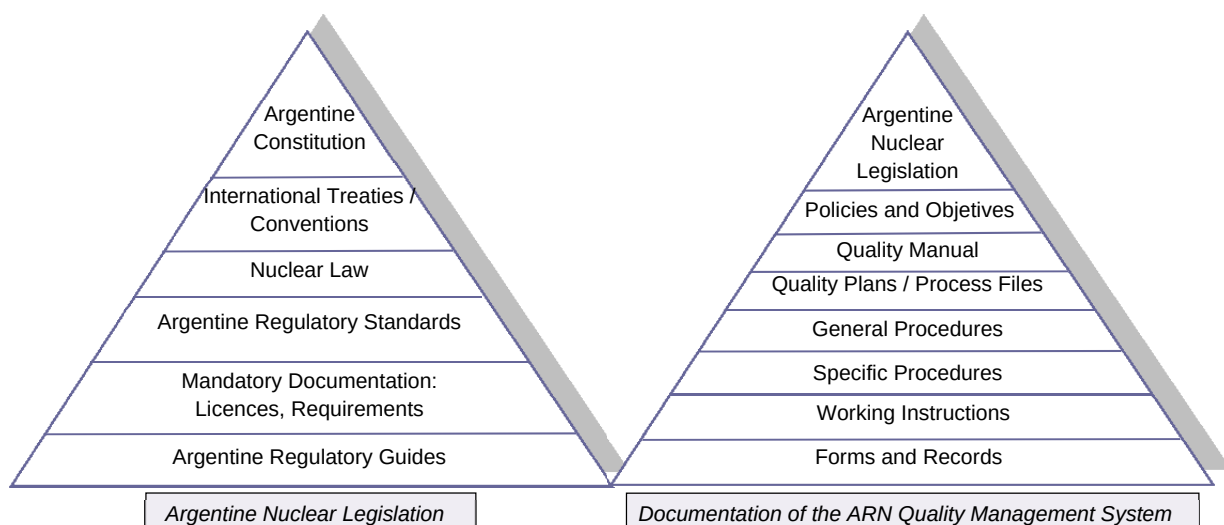
Regarding laboratory accreditation, the following assay and calibration laboratories are accredited according to ISO/IEC 17025:2005:

- 1) Environmental Control Laboratory: ARN has increased the number of accredited techniques applied in this Laboratory (in force from February 2011 to April 2015). This laboratory participates in the annual intercomparison exercises together with the National Institute of Standards and Technologies (NIST), and ensures their measurements quality through a verification program which includes the corresponding variables and the used equipment for measurement.
- 2) Biological Dosimetry Laboratory: Cytogenetic assay on human blood, applying ISO Standard 19238 "Radiation protection -- Performance criteria for service laboratories performing biological dosimetry by cytogenetic" (in force until February 2014). This is the only laboratory in Latin America and the Caribbean which is part of the Response and Assistance Network system (RANET) belonging to the IAEA to face with radiological and nuclear issues.
- 3) Physical Dosimetry Laboratory: calibrations of radiation field detectors and measurement equipment for superficial contamination (in force from 2009 to 2017).
- 4) Internal Dosimetry Laboratory: determination of iodine in thyroids was accredited in March 2013.

The respective certification and laboratory accreditation documents are available on the ARN Site: <http://www.arn.gob.ar>.

3.8.4.2. DOCUMENTATION REQUIREMENTS

The QMS is based on a solid documental structure, which involves:



The QMS documental structure is formed by external and internal documentation. The internal documents, which are generated by the different processes that make up the ARN, are updated and electronically controlled by the ARN internal Web (Intranet). The external ones, which are necessary for process development, are available through internet or information received from the ARN library. Each process is responsible for controlling and updating these documents.

Up to May 2013 there were 146 approved documents and 18 are on the approval process.

3.8.4.3. SATISFACTION OF STAKEHOLDERS

ARN focuses on the requirements of stakeholders, ensuring that safety is not compromised, while performing regulatory actions regarding laws, public and personnel safety and environmental protection matters. Satisfaction surveys suppliers were carried out regarding technical and administrative aspects related to regulatory activities.

During the periodic implementation meetings carried out by the Quality Management Unit (QMU) related with the different processes, different methodologies aimed to measure the stakeholders' satisfaction, according the needs of each process, are proposed and discussed. The objective of the above mentioned meetings is to assess the proposals to improve the quality management system.

Since 2010, many satisfaction examinations have been developed and applied to different stakeholders. They are:

- Permanent survey to licensees and other users.*
- Postgraduate course satisfaction survey to teachers and participants of the radiological protection and nuclear safety courses, twice a year.*
- Satisfaction survey for calibration and assay laboratories customers, once a year.*

In 2011, the ARN has signed the Citizen Commitment Document through which its objectives of guarantee the radiological and nuclear safety, the non-proliferation guarantees and the physical protection were strengthened. Moreover this document is a communication tool to the society, to establish some quality standards, through which the ARN assures its fulfilment and the commitment of maintaining and improving them. They are periodically monitored, measured and reported.

3.8.4.4. PLANNING

Annually, the ARN Board of Directors approves the Work and Budget Plan, which contains the activities, tasks and projects of each area, in order to comply with the Institution's objectives and goals.

The **QMS** changes are performed in a planned way, to assure its continuity, effectiveness and efficiency. *These changes are assessed through review meetings, carried out between the Board of Directors and the QMU, aimed to improve the different processes. Moreover, these meetings serve to assess the changes implementation in the processes that are being aligned to the ISO 9001.*

3.8.4.5. MEASUREMENT, ASSESSMENT AND IMPROVEMENT

ARN carries out quality self-assessments *by internal audits* in order to:

- demonstrate products and processes compliance with stakeholder's requirements.
- ensure the **QMS** compliance.
- continuously improve the effectiveness and efficiency of the **QMS**.
- identify opportunities for improvement.
- verify that corrective actions are carried out and assess their effectiveness.

The *internal audits* are performed by qualified personnel. They are independent of the area which is assessed.

During the period *2010-2012, forty eight* internal quality audits to different processes and *fifteen* external quality audits (from the Certifying Body) were carried out.

Improvement opportunities are identified by the QMU and the process responsible. Besides, quality meetings aimed to verify the application of suitable management tools and treatment of audit findings, are periodically performed.

The Board of Directors carries out annual reviews of the QMS performance aimed to assess the implementation of the identified improvement opportunities and the related changes to be made.

3.8.5. COMMUNICATIONS WITH THE STAKEHOLDERS

ARN has implemented communication mechanisms with the stakeholders concerning:

- Legislation, standards, regulatory guides, licenses, permits, authorizations.
- Feedback regarding consultations, complaints, opinions and opportunities for improvement *coming from licensees, users, unions, national authorities and organizations, among others.*
- *Participation on forums and seminars. In 2013 the ARN has organised an international seminar on nuclear policy, with participants coming from different areas such as social communicators, politicians, scientists, students, regulators and licensees, among others. This opened discussion meeting produced an opportunity for enrich and increase the communication mechanisms and tools on nuclear subjects.*

ARN regularly issues publications which inform and broadcast to stakeholders the activities undertaken by the institution.

In ARN's Website (www.arn.gob.ar), regulatory information, the Regulatory Standards, laws and acts, permits and operating licenses, annual reports; and general public communications among others, can be found.

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, Act N° 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 Act 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 2010 and 2013 are shown in Table 3.8.1. An important increase of the 2013 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 2013 is composed of: 69% from the National Treasury, 30% from annual regulatory fees and goods or resources assigned according to applicable laws and regulations, and 1% from donations.

Table 3.8.1. – ARN comparison Budget for financial years 2010 and 2013

<i>Item</i>	<i>\$ (in thousands of Argentine pesos)</i>	
	2010	2013
<i>1 - Personnel</i>	51,864	115,781
<i>2 - Support goods</i>	3,195	4,388
<i>3 - Services</i>	59,320	65,958
<i>4 - Equipment</i>	6,069	11,385
<i>5.1 - Fellowships</i>	800	1,900
<i>5.9 - Transfers</i>	6,449	11,361
<i>9 - Other expenses</i>	1,800	1,900
TOTAL	129,497	212,673

The ARN percentage Budget distribution is shown in Figure 3.8.7.

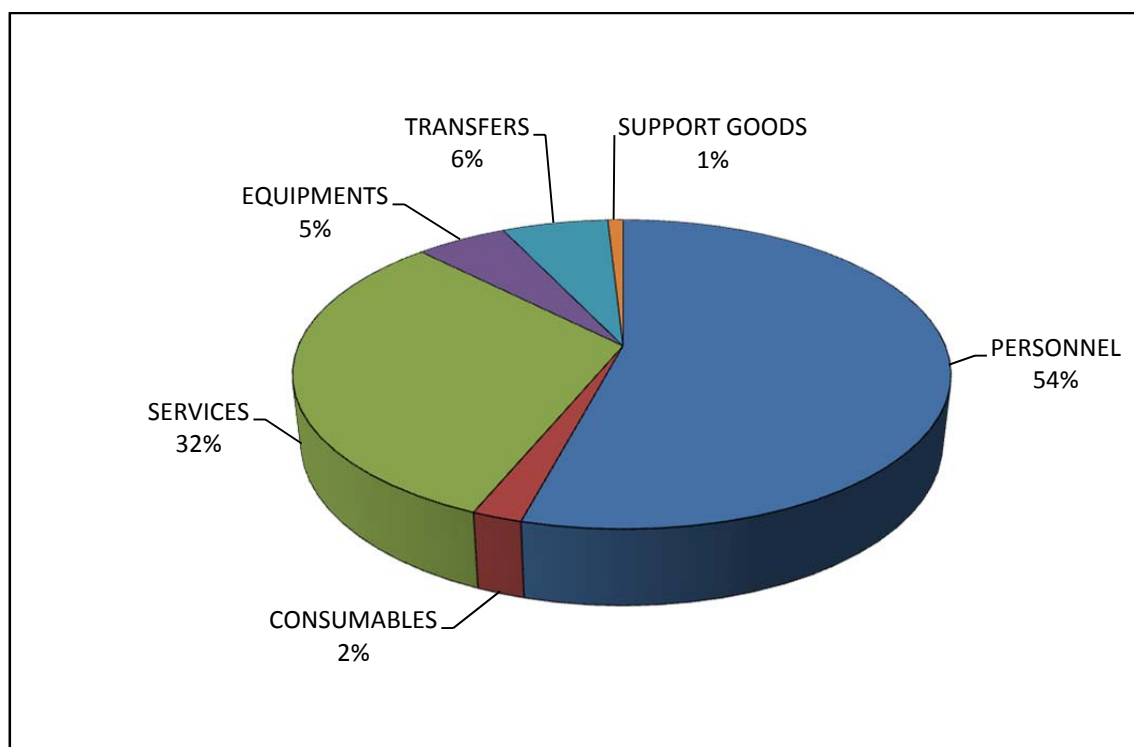


Figure 3.8.7. – ARN 2013 percentage budget distribution by category.

Additionally, the financial resources distribution for 2013, by tasks and type of inspection, is shown in Figures 3.8.8. and 3.8.9.

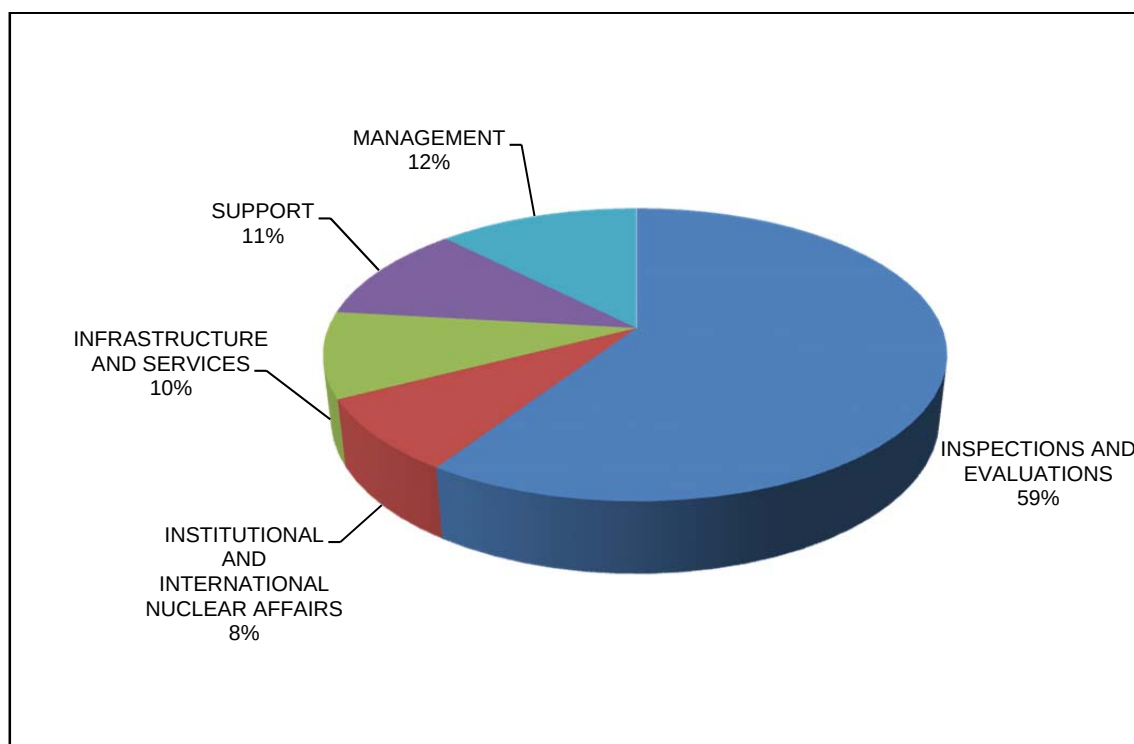


Figure 3.8.8. – Budget distribution in 2013, by tasks.

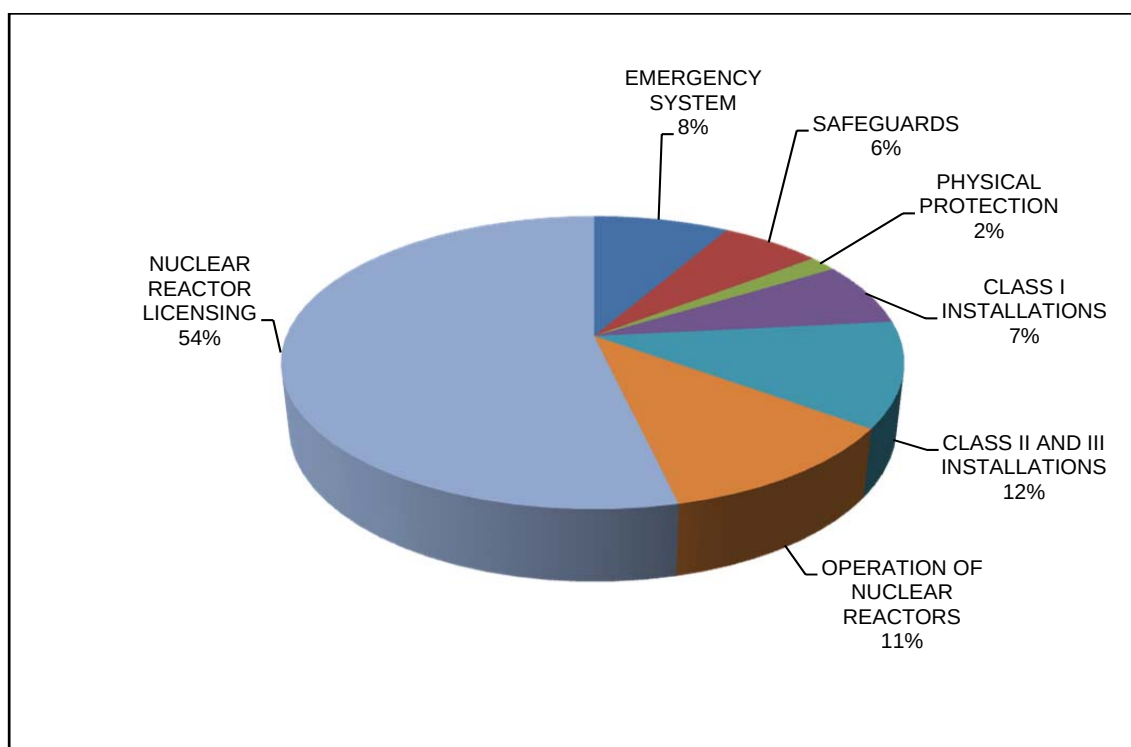


Figure 3.8.9. – Budget distribution in 2013, by type of inspection

3.8.7. RELATIONSHIP WITH OTHER ORGANIZATIONS

Regarding to its functions, ARN keeps an active interaction with several national, 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 remains the same. In this context, could be highlighted that, 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 [2010](#) until now, in the following meetings:

EVENT	PLACE	DATE
<i>Technical Committee</i>	<i>Montevideo, Uruguay</i>	<i>November, 2010</i>
<i>Technical Committee</i>	<i>Santiago de Chile, Chile</i>	<i>July, 2011</i>
<i>Plenary</i>	<i>Santiago de Chile, Chile</i>	<i>July, 2011</i>
<i>Technical Committee</i>	<i>Lima, Republic of Perú</i>	<i>November, 2011</i>
<i>Technical Committee</i>	<i>La Habana, Cuba</i>	<i>July, 2012</i>
<i>Plenary</i>	<i>La Habana, Cuba</i>	<i>July, 2012</i>
<i>Technical Committee</i>	<i>Rio de Janeiro, Brazil</i>	<i>November, 2012</i>
<i>Technical Committee</i>	<i>Madrid, Spain</i>	<i>June, 2013</i>
<i>Plenary</i>	<i>Madrid, Spain</i>	<i>June, 2013</i>

A number of agreements with domestic and foreign organizations serve as the framework for the 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
<i>General Cooperation Agreement between the Ministry of Health of the Government of the Autonomous City of Buenos Aires and the ARN. March 2012.</i>	<i>Its main purpose is to establish the basis for adoption and the development of measures for joint coordination and cooperation in areas of common interest of the signatory parties.</i>
<i>General Cooperation Agreement between the ARN and the National University of the Interior of Buenos Aires Province (Universidad Nacional del Centro de la Provincia de Buenos Aires). July 2010.</i>	<i>Its purpose is to maintain a system of reciprocal cooperation for undertaking studies, assessments, research, technology development and training of specialized personnel in the areas of Radiation Protection and Nuclear Safety.</i>
<i>General Cooperation Agreement between the Litoral University and the ARN. June 2012.</i>	<i>Develop a coupled neutronic thermohydraulic model for CNE.</i>

Table 3.8.3. – Agreements with Foreign Organizations

AGREEMENT COUNTERPARTS	AGREEMENT OBJECTIVES
<i>Arrangement between the Autoridad Regulatoria Nuclear of Argentina (ARN) and the Canadian Nuclear Safety Commission for Technical Co-operation and Exchange of Information related to the Long-Term Operation and Refurbishment of the Embalse Nuclear Power Plant. September 2011.</i>	<i>The Commission will, on request from the ARN, perform support for selected topics in the frame of: the review of the licensing process, design, construction, assembly, preliminary tests, start-up, life extension and refurbishment for the NPP.</i>
<i>Cooperation Arrangement between the National Nuclear Energy Commission (CNEN) of the Federative Republic of Brazil and the Regulatory Body (ARN) of the Argentine Republic for Import and Export of Radioactive Sources. September 2011.</i>	<i>To establish a framework for the Parties to exchange administrative information with reference to the import and export of radioactive sources.</i>
<i>Implementing Agreement on Thermal-Hydraulic Code Applications and Maintenance (CAMP) between the United States Nuclear Regulatory Commission (NRC) and the Regulatory Body (ARN). December 2011.</i>	<i>To establish a framework to work together for cooperative research in thermal hydraulic research programs sponsored by the Parties.</i>
<i>Cooperation Arrangement between United States Nuclear Regulatory Commission (NRC) and the Regulatory Body (ARN). March 2013.</i>	<i>Develop the tools for use the codes MELCOR and MAACS to be applied in safety assessments (PSA L2 and L3).</i>

Most of the agreements referred to in this and previous reports include clauses for its 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. 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).
- Nuclear Security Guides Committee (NSGC).

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

Personnel of the ARN is also frequently called by IAEA as experts for technical assistance missions to several countries, preparing safety-related publications and providing training for foreign trainees.

As were mentioned in section 3.8.3.1.4., every year ARN offers Post Graduate Courses on Radiological Protection and Nuclear Safety, which started in 1980.

3.8.8. COMPLIANCE WITH THE OBLIGATIONS IMPOSED BY THE CONVENTION

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

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

3.9. ARTICLE 9: RESPONSIBILITY OF THE LICENSEE HOLDER

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

3.9.1. BACKGROUND

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

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

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

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

Thus, when the operational characteristics of installations make it advisable, the Regulatory Body requires that those persons holding certain positions in the operation chart must receive specialized training and have their own individual license. Besides, the training requisites for the whole operation personnel have been increased (see article 3.7.).

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

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 organization known as Licensee is Nucleoeléctrica Argentina S.A. (NA-SA), company in charge of the operation of the Argentinean NPPs. The Regulatory Standards AR 0.0.1. and AR 10.1.1. establish its responsibilities, being the follows the significant ones:

- The Licensee must do whatever is reasonable and compatible with its possibilities regarding safety, fulfilling at least with standards and requirements issued by the Regulatory Body. Such responsibility extends to the stages of design, construction, commissioning, operation and decommissioning of the NPPs.
- The fulfilment of Regulatory Standards, procedures and other mandatory documentation is a necessary but not sufficient condition concerning the Licensee's responsibility, which must do whatever reasonable and compatible with its possibilities regarding safety. Besides, it must follow the standards and obligations imposed by other competent bodies not related to radiological aspects (e.g. conditions for conventional discharge of chemical effluents).
- The Licensee may support the operation of more than one installation and delegate the execution of the tasks totally or partially, but it must maintain the whole responsibility.
- In each NPP the Licensee must appoint a person of its own body, named Primary Responsible, who will be assigned the direct responsibility for the radiological and nuclear safety of the plant, as well as for the fulfilment of standards, licenses and requirements applicable to it. In the case of a NPP in operation, its plant manager is usually the respective Primary Responsible.

- The Licensee must provide the necessary support to the Primary Responsible in order to allow him to perform his task and responsibility, and must supervise him to verify that he carries out his responsibility satisfactorily.
- The Licensee must submit to the Regulatory Body the technical documents needed to evaluate the safety of the NPP which the operating license is applying for.
- No modification of a NPP related safety system design, operational features or mandatory documentation contained in the operating license, can be initiated without previous Regulatory Body authorization.
- Both the Licensee and the Primary Responsible must facilitate the performance of regulatory inspections and audits, every time the Regulatory Body requires it.
- Every change in the Licensee structure, that could affect its capability of carrying out its responsibilities, must be previously approved by the Regulatory Body.

Moreover, the Licensee must assume the civil responsibility that the Vienna Convention on Civil Responsibility for Nuclear Damages (ratified by Act Nº 17.048, 1966) has determined for the licensee. Act 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

The Licensee (NA-SA) carries out a Communication Annual Plan whose actions are intended to maintain an open and transparent communication with the stakeholders. Through the different programs, its intention is to install a positive perception of nuclear power generation. In this context, NA-SA has communicated the commissioning start-up of CNA II as well as the Life Extension project of CNE. The more relevant mechanisms applied by NA-SA are the followings:

3.9.3.1. COMMUNICATION CAMPAIGNS

Mass media campaigns are annually develop. During 2013 it was launched a campaign with the slogan "Energy in Motion". The campaign includes radio spots, television, newspapers and social networks.

During the summer season itinerant public presentations regarding nuclear energy in different tourist zones along the country are performed. This activity offers the public the possibility to learn the way how energy is produced by a nuclear manner and, also allows the public ask questions regarding the NPPs, including the corresponding safety aspects, aimed to generate a fluid and interesting dialogue.

Besides, NA-SA organizes educational campaigns in the schools that belong to localities near to CNA I, CNA II and CNE. This activity is designed with the goal to provide information regarding nuclear energy to teachers and students to strengthen links with the areas that are influenced by the facilities as well as disseminate this information among the relatives as a way of favouring dispersing it into the public.

3.9.3.2. RELATIONS WITH THE MEDIA

A strong relationship has been maintained with the media more dedicated to the energetic sector, national and local press. In this context, nuclear workshops exclusively for journalists are periodically performed. Weekly 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 oriented 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 in digital communication channels belonging to NA-SA. An active web site is available for the public and a newsletter is sent to more than 40,000 subscribers. In addition, NA-SA's communication also gathers strength through social networks such as the presence in Facebook, Twitter and YouTube.

3.9.3.4. RELATIONS WITH THE COMMUNITY

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 manner, reinforcing the knowledge related with the individual and collective performances. These presentations take place in a space where the participants can express their concerns and receive the corresponding explanation and answer by specialized personnel.

3.9.3.5. PLANT TOURS

A program of visits to NPPs aimed to the public is being carried out since many years ago. In 2012 more than 10,000 visitors were received. The general public as well as different institutions like national and international agencies representatives, schools and universities continue participating of this program.

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

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

3.9.5. COMPLIANCE WITH THE OBLIGATIONS IMPOSED BY THE CONVENTION

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

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

3.10. ARTICLE 10: PRIORITY TO SAFETY

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

3.10.1. GENERAL PRINCIPLES

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

This priority to safety is made clear in the sense that these stages must be carried out in accordance with an adequate regulatory system of principles, criteria and safety policies applied in the country for decades. Argentina, contributes to international organizations such as IAEA and ICRP on such matters.

Two of the principles the regulatory system must comply with, are:

- The regulatory control, and
- The responsibility for safety.

Two organizations are involved in the compliance with the above-mentioned principles:

- The ARN in the case of the regulatory control, and
- The Licensee in the case of the responsibility for the safety.

Due to both principles, these two institutions coexist, being completely independent one from the other.

The 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 Act N° 24.804, the rules that were consequently dictated, international agreements and non-proliferation policies adopted by the country, and
- Prevent intentional acts that may cause severe radiological consequences or the unauthorized removal of nuclear materials or other materials and equipment of nuclear interest subject to regulation and control.

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

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

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

- It complies with pertinent Regulatory Standards and requirements and performs, in addition, all what is reasonable and compatible with its possibilities in favour of radiological and nuclear safety in NPPs, concerning their design, construction, commissioning, operation and decommissioning. To that end, and according to NPPs operation, NA-SA takes the following documents into account:

- Operating License.
- Safety Analysis Report.
- Policies and Principles Manual.
- Operating Manual.
- Maintenance Manual.
- Quality Management System Manual.
- Radioprotection Procedures (Code of Practice).
- In-Service Inspection Program.
- Periodic Test Program.
- Emergency Plan.
- Personnel Qualification and Training Program.
- It improves the existing safety practices continuously.
- Ensures that those guides accepted and adopted by the nuclear industry are being fulfilled, when applicable for the case of domestic NPPs.
- It sustains an attitude towards safety based on the organization's self-evaluation, the feedback of operative experience, technological development and the early prevision of possible degradation of the plants that might affect their safety.
- It continuously carries out training and retraining courses for the plant personnel or for those members of the staff who perform safety related tasks.

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

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

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.
- Modifications to the installation and procedures (safety or safety related systems).
- Abnormal situations.
- Periodic evaluation of the installation performance.
- Periodic evaluation of the training personnel program.
- Emergency plan (including the corresponding exercises), etc.

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

On the other hand, the CRT, which is independent from the Plant Managers, advises the highest staff level of the Licensee as regards the safe operation of the NPPs, analyses the importance of the

failures, abnormal and significant events, evaluates the proposed design modifications that may affect the safety related systems, and in general about the same subjects dealt with by the CIAS.

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

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

3.10.2.2. SAFETY CULTURE AND ITS DEVELOPMENT

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

The ARN (regulatory control principle) and the Licensee (responsible for safety principles) are continuously involved in the compliance with the above mentioned principles.

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

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

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

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

3.10.2.3. COMMITMENT TO SAFETY

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

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

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

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

3.10.2.4. MANAGEMENT ATTITUDES TOWARDS SAFETY

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

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

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

3.10.2.5. PERSONNEL MOTIVATION

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

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

- Initial training,
- Periodic retraining,
- Exchange of experience (operational, maintenance, etc.),
- Professional prestige (both in their own institution and in the national or international nuclear community), and
- The preservation of the working position and chances for promotion (concerning both technical and pecuniary aspects).

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

3.10.2.6. SAFETY CULTURE REGULATORY ASSESSMENT

Evaluation of SC is inherently drawn in every inspection, both routinely and specially programmed. Examples of safety culture monitoring aspects can be seen whenever management weaknesses are detected or operations procedures are ignored or operating limits are exceeded or maintenance weaknesses are shown, as inspection findings.

Specially programmed inspections are focused on some others issues like verbal communication with operators. Besides, through evaluation of operation (routine surveillance tests, corrective maintenance) together with the evaluations of some safety performance indicator trends, it is also possible to determine signs of declining safety culture.

Safety Culture evaluation could be made as follows:

- Direct evaluation: day-to-day operation follow-up is a positive approach that is applied through the resident inspectors, whose job also includes detecting early sign of declining SC.
- Evaluation through follow-up of Operating Experience Feedback Program: Additionally, safety analysts from the ARN apply a different perspective throughout assessments. The main goal to be reached by the analysts is the evaluation of the influence of organization and management (organizational factors) as the root and direct cause of events.

The regulator/operator relationship has shown an important advance in SC. As an example, all the major findings from regulatory audits carried out since last years are well accepted by operators, and corrective actions are implemented as soon as practicable. The minutes (coming from the periodical technical meetings among regulators and operators to consider regulatory matters) that include operators commitments are met. Consequently this led to a significant reduction of formal regulatory requirements.

Additionally, an increasing number of corrective actions and design changes coming from the better application of the Operating Experience Feedback have been observed during the last years.

3.10.2.7. 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 both NPPs, IPERS mission to CNA I and OSART mission to CNE.
- Interaction with official and non-governmental bodies, with the purpose of analysing emergency preparedness measures, including the role of the ARN and other organizations (i.e. Civil Defence at national level).
- Interaction with the Uruguayan and Brazilian Regulatory Bodies in their character of neighbouring country, with regard to their participation in the practical emergency plan exercises at CNA I.

3.10.3. COMPLIANCE WITH THE OBLIGATIONS IMPOSED BY THE CONVENTION

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

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

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

3.11.1. INTRODUCTION

The Licensee of the Argentinean NPPs, Nucleoeléctrica Argentina S.A. (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 Argentine Economy Ministry.

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 N° 24065 and Law N° 24776) that guarantee all the needs for plant operation, maintenance and, the resources needed to support nuclear safety.

During the period 2010-2012 the industrial electrical tariffs followed the increase of generation costs, while residential tariffs remained unchanged.

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

The average increase of electrical energy demand in this period was 4.64% per year, while the installed capacity has an increase of approximately 15%.

This increase came from the installation of combined cycle plants through a joint investment of private capital and the National Government as well as to the hydraulics capacity increase. Also there were some small enterprises in the energy area carried out mainly by the National Government.

Referring to nuclear generation, CNA II is in the commissioning stage. Moreover, at the end of 2009 the Nuclear Activity Act was approved in which the life extension of CNE and the construction of a Fourth NPP were declared subjects of National Interest.

In Figure 3.11.1. can be finding the electrical generation from 2010 to 2012, classified in hydroelectric, thermal, renewable and nuclear.

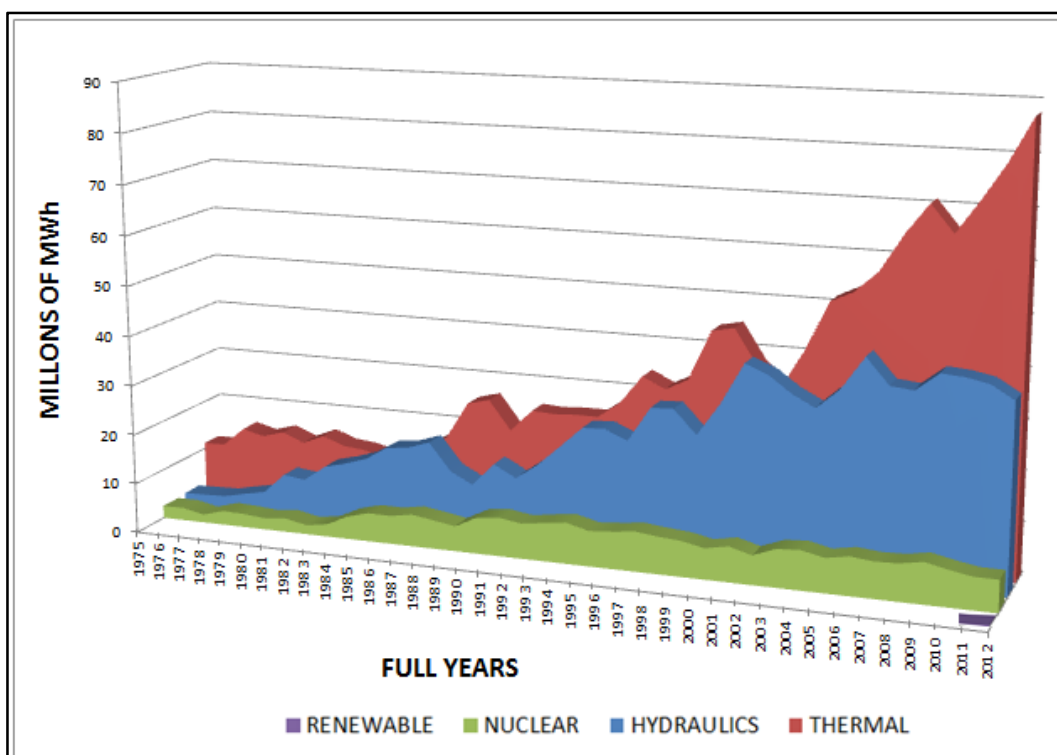


Figure 3.11.1. – Electrical generation in Argentina from 1975 to 2012

3.11.2. ELECTRICAL GENERATION AND ECONOMICS

3.11.2.1. TOTAL ELECTRICAL GENERATION PERIOD 2010 - 2012

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

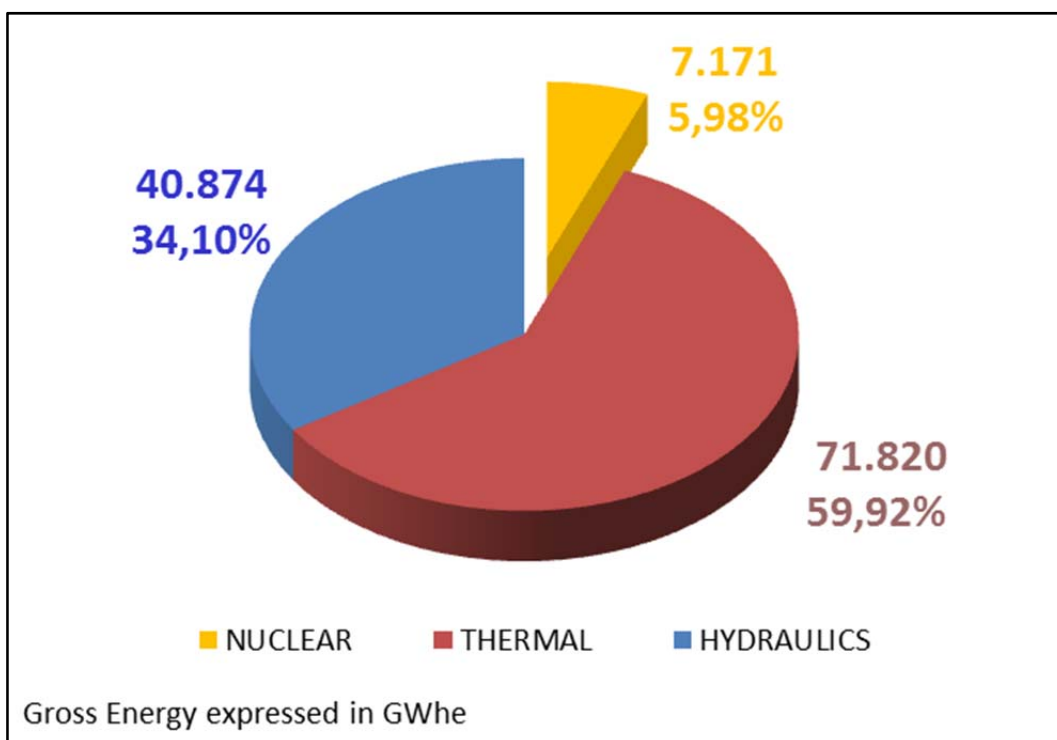


Figure 3.11.2. – Gross Energy Generated in 2010

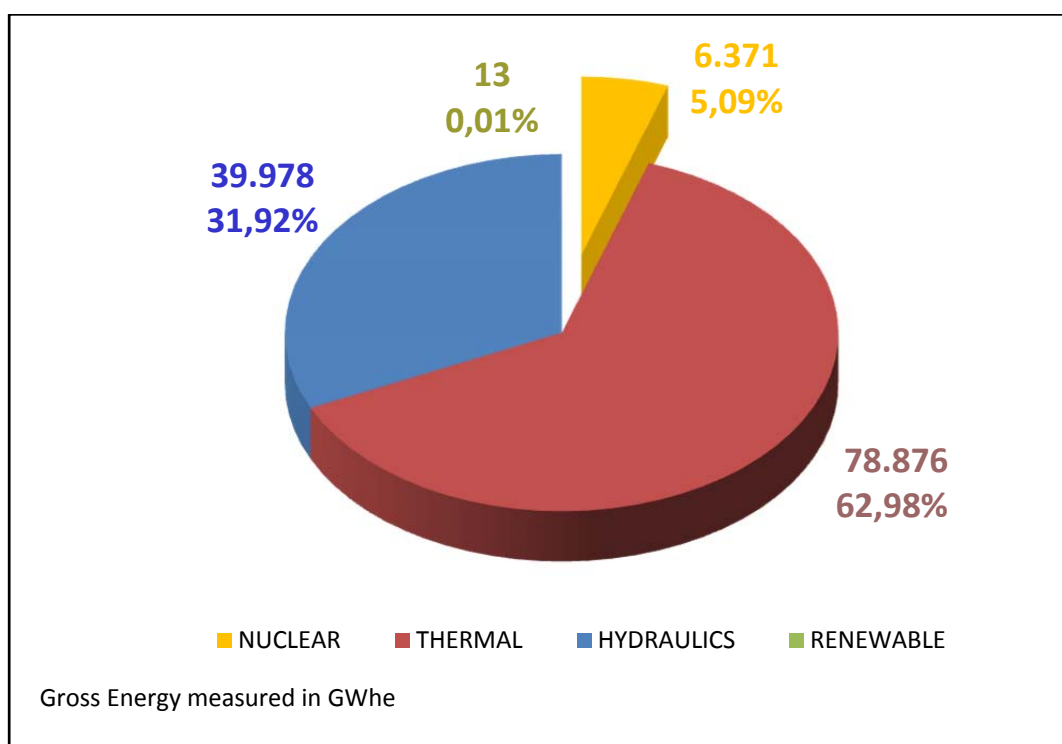


Figure 3.11.3. – Gross Energy Generated in 2011

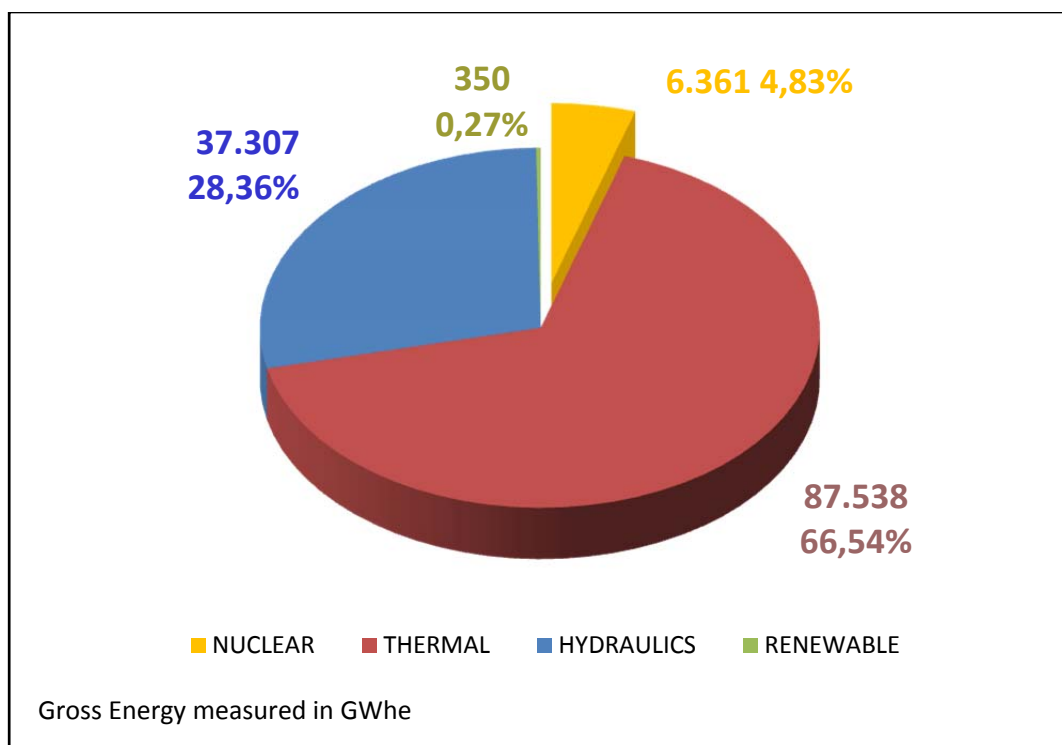


Figure 3.11.4. – Gross Energy Generated in 2012

3.11.2.2. NUCLEAR GENERATION 2010, 2011, 2012

Performance is shown in Table 3.11.1.

Table 3.11.1. – Performance

	2010	2011	2012
Gross Energy (MWh)	7,170,885	6,370,904	6,361,168
Load Factor (%)	81.45	72.37	71.86
Installed Nuclear Power (%)	3.57	3.41	3.23
Generated Nuclear Power (%)	5.98	5.09	4.83

3.11.2.3. HUMAN RESOURCES

After several years without incorporation/renewal of personnel, considering the average age of the employees and the personnel near retirement, NA-SA requested authorization to develop a 5 year program in order to incorporate new professionals. This Programme covered the period 2002-2007. Continuing that programme implementation, the process of recruitment and training of junior professionals, already mentioned in the previous report, continued. This process will continue over the coming years in order to cover the staff requirements for the commissioning and operation of CNA II.

During the period 2010-2012 the NA-SA staffs has been increased as is showed in Table 3.11.2.

Table 3.11.2. – Personnel 2010, 2011 and 2012 by work area and specific knowledge

	Year	CNA I	CNA II	CNE	Main Branch	TOTAL
Professionals	2010	181	122	219	197	597
	2011	209	120	244	194	647
	2012	258	114	260	197	715
Technicians	2010	430	83	410	27	867
	2011	499	79	492	35	1026
	2012	699	75	529	41	1269
Administrative personnel	2010	75	19	88	83	246
	2011	75	18	92	77	244
	2012	84	17	78	76	238
Total	2010	686	224	717	307	1710
	2011	783	217	828	306	1917
	2012	1041	206	867	314	2222

Considering the new projects NA-SA is involved in: conclusion and commissioning of CNA II, Life Extension of CNE and the Feasibility Assessment for a Fourth NPP, the rhythm of incorporation of new personnel had to be increased. It must be underlined that after the completion and commissioning of CNA II, NA-SA must start operating the station; therefore it must count on trained personnel. Personnel working in positions which require a special authorization from the ARN must meet strict training requirements (see article 3.7.).

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

In the particular case of CNA II, the following sources of staff recruitment and training needed to complete the project were used:

- Transfer of Technology Contract with the designer (Training and placement of personnel, transfer of design basis documentation).
- Retained and recovered capabilities (150 persons with experience in the Project and 70 more from the nuclear industry, with a total training in Siemens of 450 man-months).
- Foreign Personnel assigned to the Project.
- Cooperation Agreements (Siemens AREVA, AECL, IAEA, GRS, UNIPI, SCK, etc.).

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

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

3.11.2.4. CURRENT EXPENSES 2010, 2011, 2012

Table 3.11.3. and Figure 3.11.5. show the evolution of Operating and Maintenance costs

Table 3.11.3. – Evolution of O&M costs (pesos \$ x 1,000,000) period 2010 – 2012

TITLE / YEAR	2010	2011	2012
SALARIES	422.2	582.3	764.0
FUEL	220.9	247.7	277.6
SERVICES	90.7	126.7	133.9
SPARE PARTS AND CONSUMABLE SUPPLIES	38.7	89.3	46.5
OTHERS	27.2	29.6	31.7
ROYALTIES	15.4	16.0	22.1
SCHEDULED SHUTDOWNS	104.7	149.2	117.1
AMORTIZATION AND INTERESTS	0.0	0.0	0.0
TOTAL	919.8	1,240.8	1,392.9

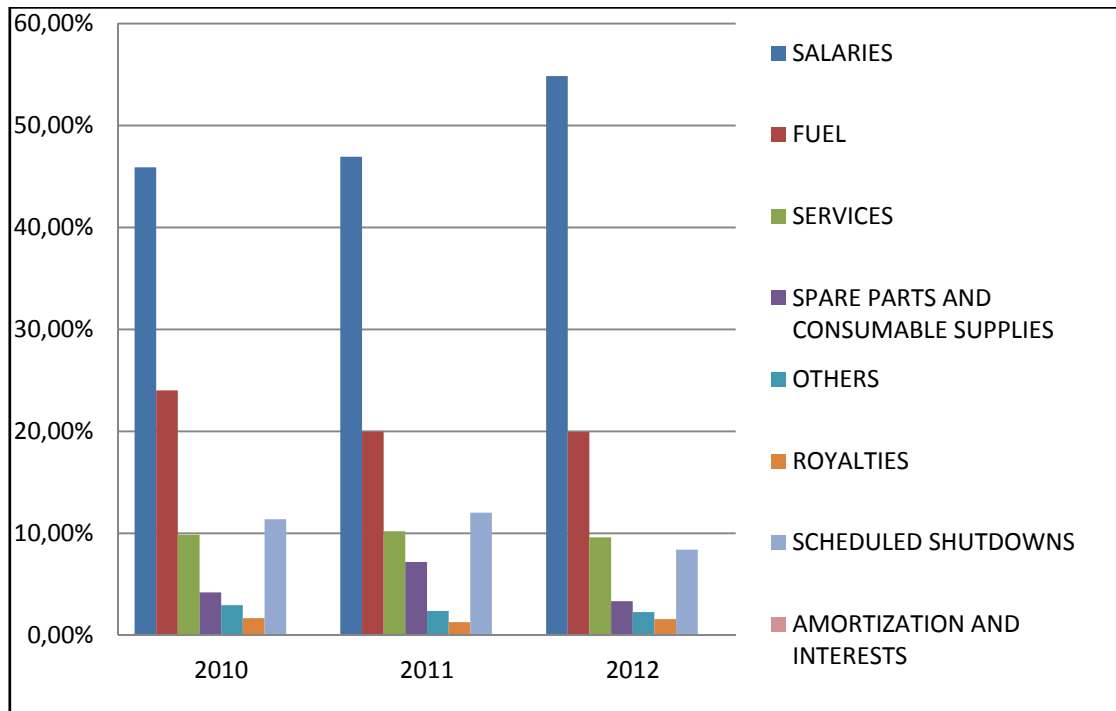


Figure 3.11.5. – Evolution of O&M costs, period 2010 – 2012

3.11.3. COMPLIANCE WITH THE OBLIGATIONS IMPOSED BY THE CONVENTION

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

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

3.12. ARTICLE 12: HUMAN FACTORS

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

3.12.1. SYSTEM TO DETECT, CORRECT AND PREVENT HUMAN ERRORS

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

In the first case, the process acts on abnormal or unexpected events that happen in the installations (operating experience, OPEX). Such events are unique opportunities to detect and as consequence to correct human errors, identifying the deficiencies regarding organization, 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.

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

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

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

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

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

Another aspect to take into account in the prevention of human errors is related to the measures taken by the Licensee regarding the contractors in order to ensure their adequate competence and safety culture. Towards this objective, the technical assistance services to the NPPs are given by competent companies whose personnel has qualifications, knowledge and experience about domestic NPPs. Also the new personnel at any level are trained with the necessary knowledge before they enter the nuclear area.

3.12.1.1. HUMAN CORRECTIVE ACTIONS IN CNA I

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

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

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

- Evaluation of low safety significant events, near misses and operational events to find the human factor related causes, in a systematic way. These evaluations are used as a learning method to prevent reoccurrence of safety significant events.
- Lessons learned from such events are used as feedback to prevent recurrence. To reach this objective, plant personnel receive specific retraining regarding the causes and consequence of each event.
- Identification of precursors to operational events. Some human factors related to low safety significant events and “near misses” can be correlated to organizational deficiencies, therefore these events may be used as leading indicators in anticipating and preventing declining performance.
- Detection of Organizational (human-related) deficiencies shows how safety must be managed to help avoiding mistakes and preventing incidents.

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

Combination of Human Performance Enhancement System (HPES) and Human Performance Investigation Program (HPIP, similar to the first one but used by regulators), are 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 were reported in Fifth National Report, it has been continued implementing a Human Performance Improving Program (HPIP), 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, and as a starting point, the behavioural expectations and good practices on the job were established and communicated through lectures and brochures.

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

1. Human Error Reduction Tools

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

- Pre-job briefing and Post-job debriefing.
- Procedure use and adherence.
- Clear communication techniques.
- Checking techniques:
 - Self-checking.
 - Peer-checking.
- Conservative decision making.
- Questioning Attitude.

The following activities *are periodically performed*:

- Spreading knowledge of the Event-Free Tools through in-hand distribution of a brochure.
- Weekly selection of one Event-Free Tool for its discussion in every meeting in the plant.

- *Contractor training on Human Error Reduction Tools.*
- Self-checking evaluation on the application of the *Human Error Reduction Tools during programmed outages.*

2. Operational Experience Application

Internal Operational Experience is used within the HPIP to recognize human conduct problems and behaviour enhancements. An annual trend analysis is carried out to assess those events whose direct causes and/or root causes or causal factors are due to human error.

Taking into account that one application of operational experience is its discussion during pre-job briefings; its insertion into work packages is considered part of this program.

3. Task Observation

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

- Survey conducts on the field.
- Help to improve the performance of the workers.
- Strengthen the good practices surveyed.
- Strengthen the plant's standard and expectations.
- To ease communication between chief-staff and job executors.

A semi-annual analysis on surveyed trends is carried out to assess the degree of compliance of the established performance expectations and to generate the needed corrective actions.

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

The expectation regarding the number of observations required per manager and supervisor was clearly defined by the Station Manager. Changes in the Task Observation form were also included.

3.12.1.2. HUMAN CORRECTIVE ACTIONS IN CNA II

CNA II is in the commissioning stage. Relevant steps have been carried out to improve human performance at CNA II.

An effort has been made in order to train the commissioning and operations staff on issues related to human factors, in order to encourage a safety culture both at the commissioning stage and the following plant commercial operation stage. In this sense, the following seminars were carried out:

- *Safety Culture. May 2008, February 2010, February – March 2011.*
- *Effective Communication Techniques (including as an item the three way communication). July – August 2011.*
- *Implementation of Operative Experience Program. July – August 2011.*
- *Error Prevention Techniques. March – April 2012.*
- *Root Cause Analysis. September 2012.*

CNA II has a series of Operating Instructions (IO) whose application helps to preserve the safety and physical integrity of individuals, as well as equipment and systems. Also, these instructions favor the detection, correction and prevention of human errors. The following are relevant examples:

- *IO-19 "Operating Experience Program during Phase A and B of the commissioning of CNA II":*

Since June 2011, this instruction has defined an appropriate methodology to the commissioning process (phases A and B) of CNA II related to report, analyse and document systematically the treatment of internal and external events linked.

The Operating Experience Program (OEP) has helped improve the safety culture and reliability of operation of plant systems and equipment.

The events occurred have been well documented in order to implement measures to prevent that these or others with similar cause happen again.

OEP implementation has created an open and critical attitude of the UG commissioning staff, as well as the CNA II Operation and Maintenance Management staff, since each event considered relevant is duly reported and analyzed, and these analyses are disseminated. In turn, a monitoring is performed to make sure that corrective actions are implemented and lessons learned are exploited.

- IO-27 Pre-Job Analysis, Pre-Work Meeting and Post-Job Review.

This instruction has entered in effect and begun to be implemented since March 2012. It sets the method for implementing error prevention tools during commissioning testing of equipment and/or systems of CNA II.

These tools are:

- *Pre-Job Analysis during tasks and tests planning.*
- *Pre-Job Meeting just before testing begins.*
- *Post-Work Review in some cases completed after complex tests, when what was performed differs to what was planned or when the tests have been successful and it is considered relevant to disseminate the lessons learned.*

- IO-02 Procedure of Preliminary Work Permit.

The purpose of this instruction is to promote safety during maintenance, mounting terminations, design changes, or any other work on SSCs transferred to commissioning.

A work permit (WP) is demanded, in order to reduce the risk or allow carrying out the task, when it is required to disable components, blocking of valves, configure the system, take special measures or perform functional tests. This WP is not required when there is no risk from the process or system, such as for visual inspection, mounting tracking tasks, etc.

When generating a WP, one Local Security Officer checks in situ work hazards regarding industrial safety and fire protection, likewise the Head of Security and Radioprotection controls that the risks associated with the task are assessed and that security measures defined by the Local Security Officer are correct.

- IO-20 Optimization of Commissioning risk jobs.

This instruction complements the instruction described above. A Risk Work Order (RWO) is attached to all WPs, Failure Notice or Maintenance Order when is required to include significant security measures to be implemented by the responsible for its execution.

When generating a RWO, a Security Technician is responsible for the safety supervision of the task assigned, who attends in the field together with the Responsible of Task Execution by analysing and defining the risks of each stage, to identify protections, to verify its implementation and, in general, to fulfil their role of taking action to make the work be safe.

- IO-22 Foreign Material Exclusion in CNA II Systems.

This instruction became effective in September 2011, encouraging a preventive attitude regarding care among the people involved in activities such as construction, mounting, commissioning, operation and maintenance.

This means that it has been accomplished that these people can determine which tasks are to be run in an area around a system or open component through which any foreign material can enter through openings in the piping and take every precaution to prevent inadvertent introduction of foreign materials.

3.12.1.3. HUMAN CORRECTIVE ACTIONS IN CNE

In the last period, some activities continue being carried out to improve human performance at CNE; some of them were the following:

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.

Thus, a Team of Human Performance was created with the following purposes:

- To raise the awareness of all Embalse NPP staff about the use of Event – Free Tools.
- To reduce the rate of events caused by human error.
- To identify and integrate processes which encourage the improvement of human performance.

The following activities were carried out with the purpose to achieve the proposed goals:

- Development, communication and implementation of a Human Performance Program.
- The Human Performance Program comprises the following activities:
 - Raise awareness in the whole staff by means of attendance to seminars given on the subject of Human Performance and Event – Free Tools.
 - Presentation of eight Event – Free Tools (Pre-job briefing and Post-job debriefing, Peer-checking, Self-checking, Independent checking, Procedures use and adherence, Three-way communication, Questioning attitude).

During the reported period have been considered the following tools:

- Pre-job briefing.
- Peer-checking.
- Three-way communication.
- Procedure use and adherence.

The learning was carried out by practical retraining in the facilities of the Training Division. This activity was done by Coordinators, Department Chiefs, Division Chiefs and Supervisors. The purpose of this methodology is to show the workers the commitment and involvement of chief-staff in the improvement of human performance.

Advertisements were placed in strategic locations and personal cards were handed out to keep in the desk or toolbox, in which are depicted the four human performance tools. The staff was also handed a daily-use handbook on subjects such as: when, how and why the human performance tools must be used, Human Performance personal adviser, Behavioural expectations and rules, Basic safety rules, and Handbook of Expectations during an outage.

In order to enhance the use of Event – Free Tools, a one-week discussion period for each Event – Free Tool and Embalse NPP process was established. Every meeting must begin with a discussion on the importance of the use of that week's given tool or process. Besides that, the information of the tool is advertised on the displays of Embalse NPP.

Embalse NPP developed the following procedures:

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

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.

In order to gather information aimed at the continuous improvement, the Human Performance Program was included in the Self-evaluation Program for CNE.

WANO provided assistance through its “Technical Support Mission Programs” on the following subjects:

- Root cause.
- Foreign Material Exclusion.
- Pre-job briefing and post-job debriefing.
- Material Conditions.
- Work Management.
- Coaching in the field.

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

- Rounds specified in the Management Manual.
- Operative Experience.
- Job programming and preparation.
- Job watching.
- Self-evaluation process.
- Condition monitoring of materials of equipment, components, and structures process.
- Foreign Material Exclusion process.
- Operational Decision Making.

The program’s effectiveness feedback is obtained from the conclusions arisen for the different processes established at CNE. Information from self-evaluations, job watching, 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.

3.12.2. MANAGEMENT AND INSTITUTIONAL ASPECTS

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

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

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

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

- Weaknesses in the safety policies.
- Weaknesses in accident analyses.
- Procedure violation.
- Operator errors.
- Deficient training.
- Deficiencies in the use of operational experience.
- Weaknesses in emergency planning.

3.12.3. HUMAN RELIABILITY ANALYSIS

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

The 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.

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

3.12.4. PROGRESS IN HUMAN FACTORS

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

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

3.12.5. COMPLIANCE WITH THE OBLIGATIONS IMPOSED BY THE CONVENTION

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

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

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

3.13. ARTICLE 13: QUALITY ASSURANCE

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

3.13.1. INTRODUCTION

The application of proper Quality Assurance Programs (QA Programs) in design, construction, commissioning, operation and decommissioning stages of nuclear installations is a regulatory requirement in Argentina. In the NPPs' case, Regulatory Standard AR 3.6.1. sets the requirements to be fulfilled. In addition, Regulatory Standard AR 3.7.1. determines when the Licensee must present the QA Program and the QA Manual.

As was stated in previous reports, based on the regulations, Argentine NPPs in operation or under construction have QA Programs that are documented, implemented, revised and evaluated by the Plant Management. The frame of these specific programs is the General Quality Assurance Program of the Licensee.

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

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

3.13.2. QUALITY ASSURANCE PROGRAM OF THE LICENSEE

Since the creation of NA-SA, the convenience arose of having a General QA Program that would be the reference frame for the specific Quality Programs of each organizational unit (NPP).

The QA General Manual describes the Quality System implemented in NA-SA. 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 NA-SA, as well as coordinate and integrate the common objectives, the mission and activities of the organization.

The QA General Manual comprises the Quality Policy for the entire organization (see [Annex VII](#)). Both NPPs in operation (CNA I and CNE) develop their own programs according to the requirements set in the General Manual.

NA-SA's QA General Program is documented in the QA Manual, together with the procedures, general documents, QA Manuals and internal procedures of the organizational units.

QA Management is responsible for evaluating the implementation of the QA Program in CNA I, CNA II and CNE. The results of these evaluations are reported to the highest level of NA-SA

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

Figure 3.13.1. shows the chart organization for QA Management.

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

ORGANIZATION UNIT	DOCUMENT	NUMBER OF PROCEDURES
NA-SA	<i>Quality Assurance Manual - Approved December 2010</i>	20
CNA I	<i>Quality Assurance Manual for the Operation of CNA I - II Approved December 2012</i>	234
CNE	<i>Quality Assurance Manual for the Operation of CNE - Approved January 2013</i>	646
UG-CNA II	<i>QAP 115 - Atucha II Project Management System Manual - Approved March 2012</i>	160
Maintenance Support	<i>Management System Manual - Approved November 2011</i>	325
Embalse NPP Life Extension Project	<i>Management System Manual - Approved May 2011</i>	130

QA Programs for NA-SA, CNA I, CNA II and CNE fulfil the requirements of Regulatory Standard AR 3.6.1., IAEA Practice Code 50-C-Q, and all IAEA applicable Safety Guides.

The QA Manual for the construction of CNA II was first issued following IAEA Practice Code 50-C/SG-Q, ISO Standard 9001:2000, and Regulatory Standard AR 3.6.1. A later upgrading comprises GS-R-3, IAEA 2006, Management system for facilities and activities; Occupational Health and Safety OHSAS 18000 (IRAM 3800 - Safety and Occupational Health) and ISO Standard 14000, Environmental Management System.

Annex VII shows NA-SA's Quality Policy and its Environmental Policy.

The Licensee verifies that every person and every 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.

NA-SA developed and implemented an Environmental Management System that fulfills the requirements of ISO 14001 standards. Its scope comprises electric generation from CNA I; electric generation and Cobalt 60 production in CNE; the maintenance, erection and commissioning of CNA II; activities related to CNE Life Extension Project; administrative activities related to the Fourth NPP Project and the technical and administrative management of the Central Office. The Environmental Management System was first certified in August 2003. ISO 14001 certificate is valid until August 2015.

Changes performed in the QA organization have allowed a better coordination of the audits performed from the Central Offices and from each NPP. These changes have also reduced the period for answering the audits and consequently the implementation period for corrective actions.

The Environmental Management System implementation significantly improved waste control, especially hazardous non-radioactive waste control.

There are some changes in NA-SA Organizational Chart with respect to those informed in the previous National Report. Figures 3.13.2., 3.13.3., 3.13.4. and 3.13.5. shows the Organizational Charts of NA-SA, CNE-PLEX, CNE and Atucha site (CNA I and CNA II) respectively.

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

3.13.2.1. QUALITY SUPERVISION OF EXTERNAL SUPPLIERS RELATED TO REFURBISHING AND CONSTRUCTION OF NPPS.

All new SSCs acquired for the plants' 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

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

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

CNA II

In order to describe this subject, Atucha II NPP contractors and suppliers; two great stages need to be considered.

- *Contractors and Suppliers during the first stage (until 2005).*

The principal contractors for the construction and erection were KWU and the Engineering Enterprise ENACE. During the period between 1995 and 2005, the acting organization was the Conservation Management of CNA II, which depended on NA-SA.

- *The selection was performed by KWU or ENACE.*
- *For the national suppliers the document "Requirements for Contractors and Suppliers" as well as the series ISO 9000 were used.*

- *Contractors and suppliers during the second stage (2006 onwards).*

The organization for the construction, erection, commissioning and operation is the CNA II Management Unit.

The guidelines for this assessment are established in the QA Procedure "OP-06-04 – Assessment of Contractors and suppliers". Among the general considerations for the selection the maximum national capacity participation was required.

CNE PLEX

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

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

- *IMPESA has completed its qualification process and has obtained the ASME qualification to manufacture class I nuclear components. The manufacture of the 4 new SGs for CNE is in progress.*
- *FAE (Special Alloys Manufacturing) was certified by ASME and has finished making all the tubes of the SGs.*
- *CONUAR has obtained the certification to manufacture class I nuclear components and the CANDU designer qualification to manufacture each component of the reactor (end fitting, locking cap, shielding cap, calandria tubes, and pressure tubes).*

3.13.3. COMPLIANCE WITH THE OBLIGATIONS IMPOSED BY THE CONVENTION

Adequate QA Programs for those activities which are important for safety throughout the nuclear installations life have been implemented for the Licensee. Therefore, Argentina meets the obligations set in Article 13 of the Convention on Nuclear Safety.

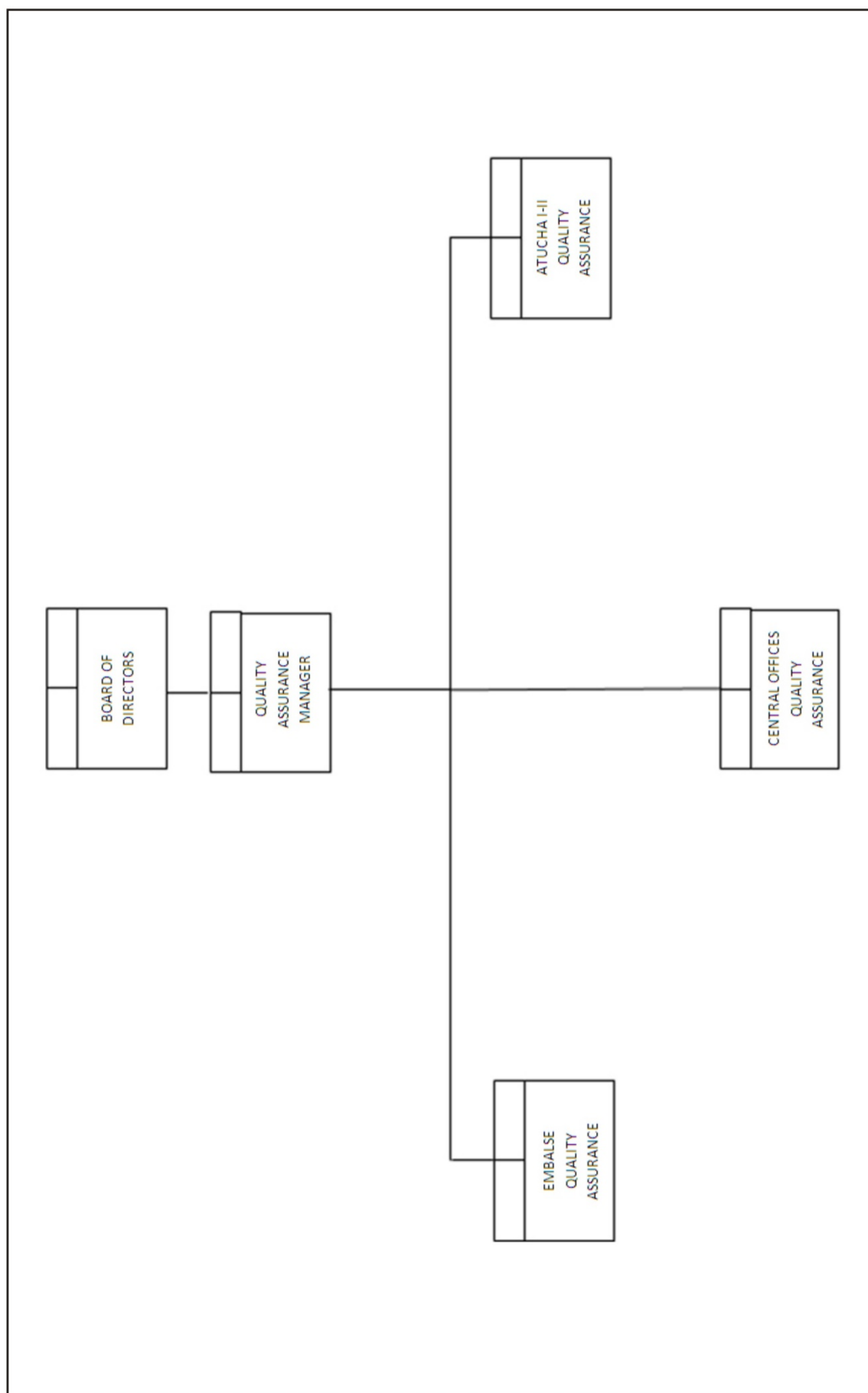


Figure 3.13.1. - Quality Assurance Division Chart

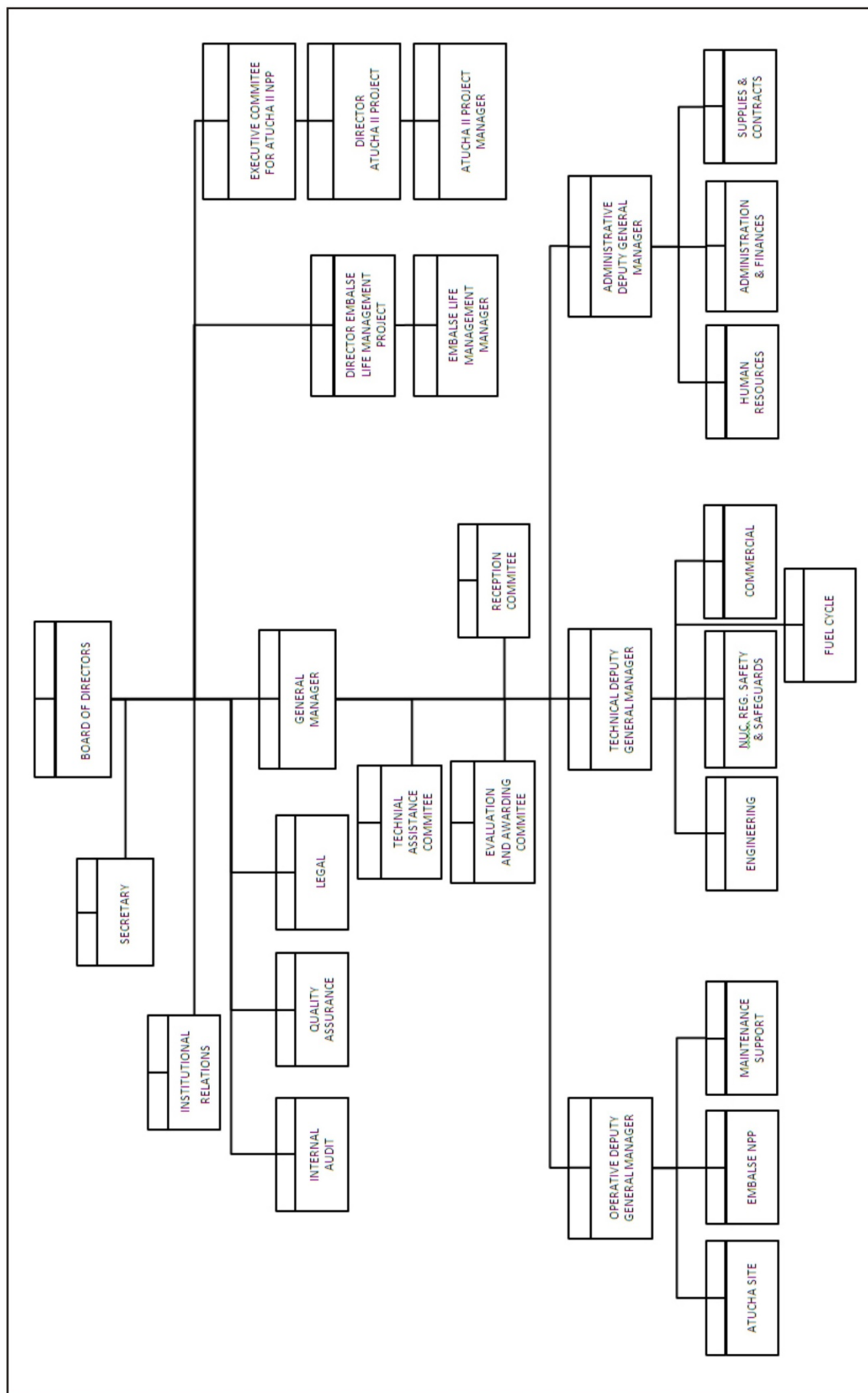


Figure 3.13.2. - Nucleoeléctrica Argentina S.A. General Organization Chart

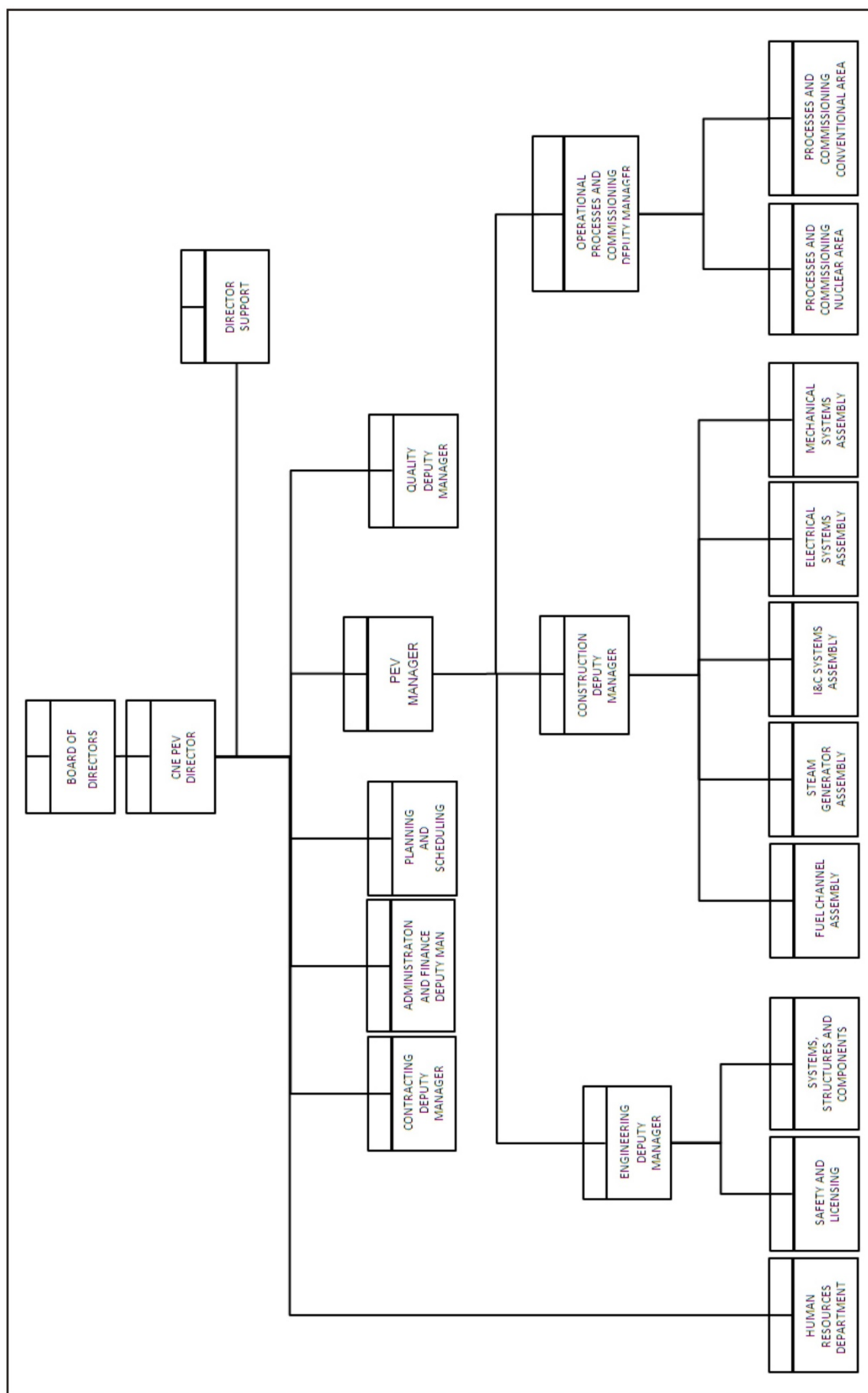


Figure 3.13.3. - CNE - PLEX Organization Chart

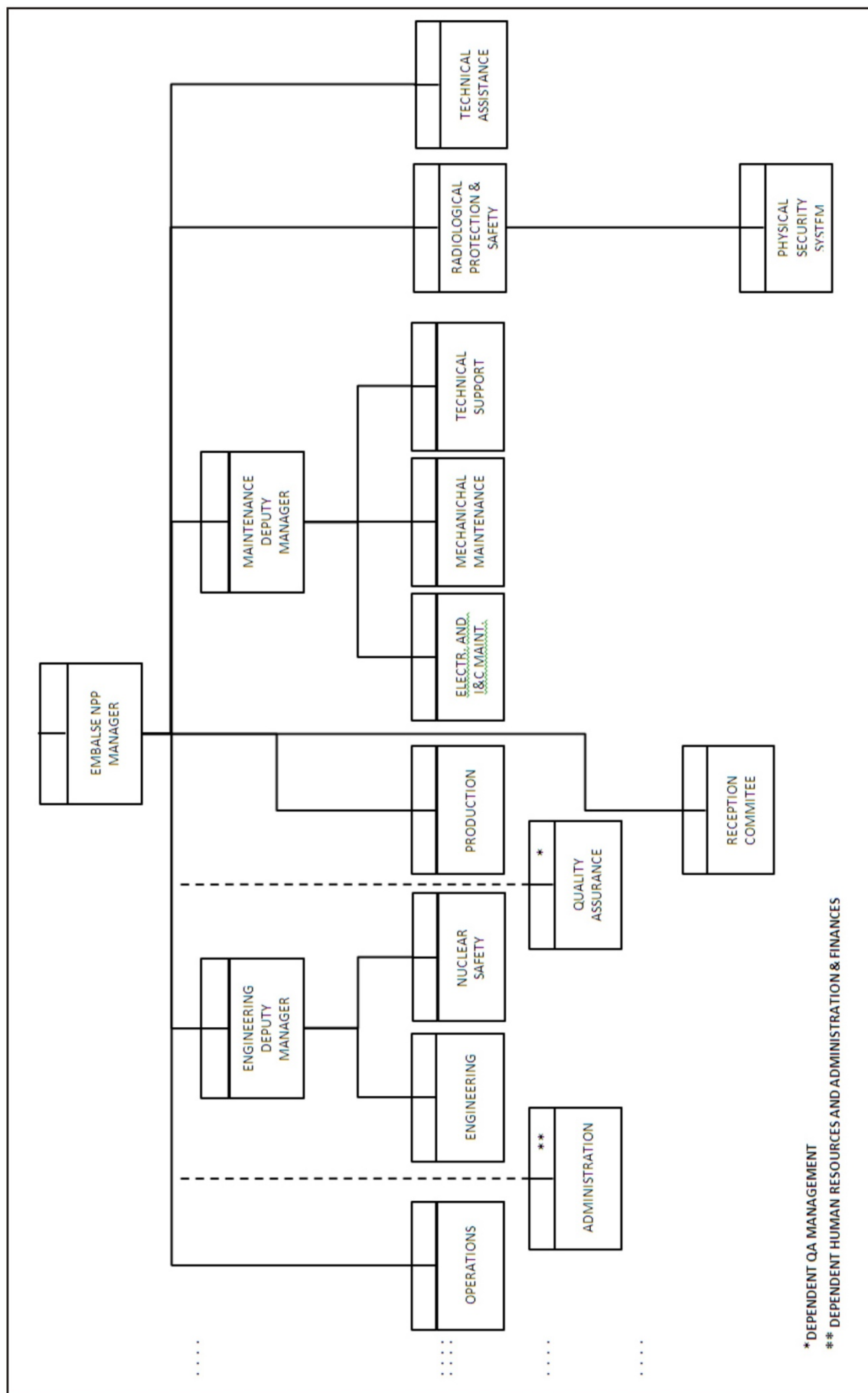


Figure 3.13.4. - Embalse Nuclear Power Plant Chart

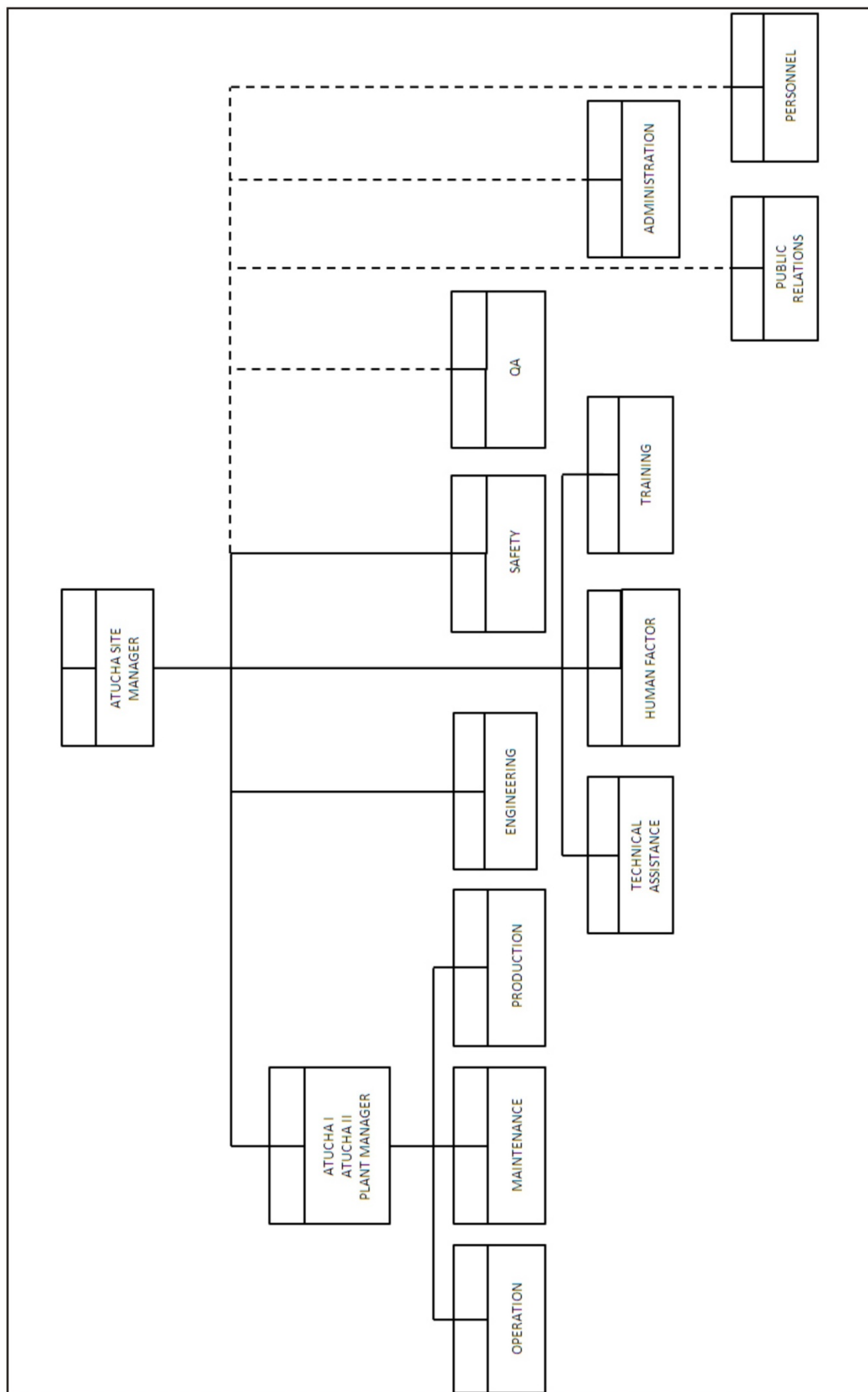


Figure 3.13.5. - Atucha Site Organization Chart

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 the information emerging from abnormal events or what is learnt through accumulated experience in the case of the assessments carried out by the Licensee. Information about abnormal events and accumulated experience comes from the NPPs itself and other domestic and foreign installations.

The above mentioned safety assessments involve 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 carried out by the Licensee. In this respect, the Regulatory Standard AR 3.7.1. sets the rules for the submission of the preliminary and final versions of this report, during the licensing process. Its periodic updating, including all the modifications performed to the installation as well as its safety improvements, is established in the Operating License.

On the other hand, the licensing process begins several years before the NPP commissioning. First of all, pre-operational studies are carried out 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, evaluate the radiological consequences of those accidents postulated in the safety analysis, elaborate an emergency plan and 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 carried out in order to analyse the impact on safety caused by modifications in the installations, by the occurrence of significant events or by any other justified reason.

For each of the existing NPP, the safety assessments carried out during the design, construction, commissioning and operation stages up to 2009 were summarised in previous Nuclear Safety Convention reports.

The main evaluations done during the period corresponding to this Convention are shown in the following Sections.

3.14.2. AGEING

The Ageing Management Program (AMP) is useful in preventing and detecting systematically any degradation that involves equipment, systems and components by affecting the design safety margins.

Each NPP in operation has implemented its own AMP, required by ARN with the objective of maintaining the safety of the NPPs' lifetime by optimising the inspection and maintenance programs, as well as, the monitoring, prevention and mitigation of ageing

The monitoring of components degradation is included within the AMP. It is based on evaluation of the operating conditions for each component under analysis, such as working temperature, irradiation field, component functional characteristics and human errors that can lead to worst normal degradation conditions.

The AMP includes the following activities:

- Safety relevant components selection to evaluate ageing effects.
- To study the selected components ageing mechanisms, including the identification / development of practical methods for ageing control.
- Remaining lifetime evaluation and management of the degradation due to ageing through surveillance, maintenance and operation by suggesting mitigation actions.

Well known methodologies are used to evaluate such degradations:

- Predictive and preventive maintenance.
- In-service Inspection program.
- Additional testing.
- Degradation evaluation.
- Evaluation of acceptance criteria in each case.

The Licensee have included the AMP within a Life Management (PLIM) Program to analyse the extent current NPP operation for both CNA I and CNE, considering that the safety of a NPP is a necessary but not sufficient condition for life extension (PLEX). These programs consist in an integration of ageing management and economic planning to maintain the safety level and maximize return of investment over the service life of the plants (*see section 3.6.5.*).

CNE has an In-Service Inspection and Preventive Maintenance Program that has been carried out during the first cycle of operation, by means of which a follow-up of the pipes / heat exchangers and components was performed, in order to prevent effects related to their use. During the PLEX Phase I, state evaluations of SSCs were carried out, where several topics were systematically analysed: the operational history, the degradation phenomena, and the type of preventive maintenance that these SSCs are subjected to. From these analyses, a set of recommendations emerged to modify both the preventive maintenance and/or the ISI plan to be implemented during the second operation cycle (extended life).

A Workshop on Concrete Degradation Management in NPPs sponsored by IAEA was held in Argentina (4 - 6 June, 2013).

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

CNA I

- *In 2010 has been created an Ageing Management Department. This department is responsible to perform the condition assessments (CAs) of safety-related SSCs, and to identify degradation mechanisms as well as take the necessary measurements to stop or mitigate the ageing effects. The department follows the guidelines adopted internationally from IAEA (NS-G-2.12) and has developed a procedure for both plants, "Integrated Ageing Management Program CNA I-II". In order to apply this procedure it was necessary to define the systems that must be included within the scope from now on. There are fourteen systems identified in CNA I including cable trays.*
- *In 2011 there was performed a plant walkdown during which photographs of all the components of interest were taken.*
- *Under the program specific procedures for evaluation of each group of SSCs are being developed. Currently, a procedure of cables' ageing is still under discussion.*
- *Courses on materials degradation.*

- *Schedule state assessments definition: some systems were already evaluated and the corresponding reports were issued.*
- *Easy fixes to improve electrical connections.*

CNA II

- *Since the components have been stored for a long period, each one of the equipment has been tested before mounting it. In addition, assessments to safety-related systems are being performed in order to determine its state before the commissioning and use this information as "reference data", i.e. as the initial state of operation.*
- *The generic Ageing Management Manual is being finished.*
- *The prioritization list of the SSCs' is being compiled.*
- *The SSCs selection to be included in the AMP for the prioritized systems.*
- *The aging management of wires was implemented.*

CNE

- *An internal technical support meeting was carried out in 2012 among specialized personnel belonging to the Ageing Management Group of CNA I and the CNE personnel involved in activities related to ageing management with the objective of exchange experiences to create an Ageing Management Department in CNE. As a meeting main result can be mentioned that was verified that many of the activities required to implement an AMP have already been conducted during Phase I of the PLEX project.*
- *Based on the ISI Program and the Preventive Maintenance System in effect on the installation, complemented with Phase I (state assessments of the PLEX project) and its recommendations, the AMP of the second cycle of CNE operation will be elaborated. For the preparation of this program will be taken into account guidelines internationally adopted, particularly those included in the documentation for the IAEA (NS-G-2.12 Ageing Management for NPPs).*
- *The cable ageing problem led to a specific program to carry out an ageing monitoring of electrical wiring. This document establishes the general criteria to determine the cables current status and to determine the degradation evolution over time. Currently, five procedures related to the program are in issuing process. These procedures, among other things, are focussed on the selection of representative samples of cables to be monitored, classification of different environments and electrical and mechanical tests to be performed.*

The Regulatory Body is checking the results of this program in a regular basis in order to review whether it is being effectively managed considering its objectives.

3.14.2.1. CNA I REACTOR PRESSURE VESSEL

CNA I started commercial operation in 1974. The base material of the RPV is similar to those of other NPPs RPV at that time, low alloy ferric carbon steel equivalent to DIN 22NiMoCr37 and similar to ASTM A 508 class 2 forging.

The companies that participated in manufacturing the RPV were the following:

- *Manufacturing of the steel base material, forging and vertical welding: Rheinstahl Heinrichshüte.*
- *RDM participated only in some welding.*

The acceptance tests and essays performed in the manufacturing process were the following:

- *Ultra sonic testing, SIEMENS procedure.*
- *Tensile tests: Charpy V notch probes 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.

RPV's inspections are included in the ISI Program and are performed according to ASME XI code. No relevant indications were reported.

Although the results showed that the RPV should operate safely until EOL, the Regulatory Body required 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, experts and international consultants.

The studies conclude that the integrity of the CNA I's RPV is guaranteed until EOL and 1.5 EOL. Also could be mentioned that the acceptance criteria as defined by the French and German guides were also applied to the results of the irradiation programs and the same conclusions were obtained.

ARN considers that, according to the evaluations which have been performed so far, the RPV integrity with respect to the material properties is assured at least until the plant design EOL. Nevertheless, the RPV integrity assessment is considered a permanent subject, for that reason the ARN considers necessary a permanent evaluation in the areas of non-destructive examinations, PTS and neutronic, 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.

Since 2010, the licensee's activities were focused on the non-destructive examination in compliance with the ISI program and with the main purpose to confirm the validity of the integrity assessment.

The areas and frequency of inspections are determined in the ISI program which is based on ASME code Section XI and, following this, during the 2011 scheduled outage, the licensee has inspected a group of bolts of the closure head. Besides, during the 2013 scheduled outage, the welds of the RPV's nozzles to pipes as well as the corresponding base material (as described below) were inspected.

In spite that the definition of the inspection areas follows the ASME code rules, considering the experience gained with the events occurred at Doel 3 NPP and Thiangé 2 NPP (flaw indications detected), the ARN and the licensee agreed to enlarge the examination volume corresponding to ASME Code to that area where quasi-laminar indications were found in the above mentioned NPPs. This experience yield the decision to inspect the base material in the upper and lower core shells.

During the last scheduled outage (April-May 2013), the inspection was made by automated ultrasonic and carried out jointly by Doosan Power Systems and the licensee and covered all plates in the RPV cylindrical proportion, no indications were detected. The examination volume and the defect acceptance standard have been defined by the licensee following the indications found in the Belgium plants:

- *100% examination of the full wall thickness.*
- *Inspection of six of vertical segments of the RPV in a full length and 700mm wide.*
- *Target flaw size: 10 mm x 10 mm quasi lamination defects are oriented parallel to the inner and outer surfaces of the RPV.*

The results of the inspections showed that no unacceptable defects have been detected within the inspection volumes of any of the six segments of base material.

On this subject an IAEA Workshop on Improved RPV Structural Integrity Assessment for CNA I and CNA II were held during March 2013 in Buenos Aires.

3.14.2.2. CNE PRESSURE TUBES

Repositions, inspections and scrapping are carried out on the Pressure tubes (PT) of CNE in order to verify that the risk of PT integrity degradation is minimum, and the PT can reach EOL in a safe manner.

3.14.2.2.1. Garter Springs Repositioning between Pressure Tubes and Calandria Tubes

The objective of garter springs repositioning between PT and Calandria Tubes (CT) is to minimize the risk of degradation of the integrity of PT as a consequence of the PT / PC contact.

The task is carried out during scheduled outages, and consists on the repositioning of the garter springs belonging to a group of fuel channels according to a given priority, in order to:

- Avoid the contact of the PT with the corresponding CT, before their EOL,
- Eliminating existing PT/CT contacts, and
- To verify, in the case that the PT/CT contact has not been able to be eliminated, that the PT does not reach the content of equivalent hydrogen for the blister formation threshold (BFT) before the next scheduled inspection.

The technical criteria for repositioning used by NA-SA is based on the variation of the deuterium uptake along the PT, taking into account the possibility given by the mechanism of susceptibility to blister formation, allowing that the cold end of the PT may be in contact with CT. This alternative requires a closer control of the deuterium intake rate.

The SLARADE computer code is used to define the repositioning strategy, and the MACACO program is used to obtain the value of pressure to give the tool for garter spring liberation. NA-SA keeps a close relationship with CANDU Energy (ex AECL) engineering and continues incorporating all the operating experience of the Canadian plants, obtaining a better estimation of the time window for PT / PC contact.

A probabilistic assessment of PT / CT contact and the potential for blister formation that could eventually lead to PT failure was performed during the 2011 scheduled outage. The probabilistic approach was selected because provides a basis for a risk-based fuel channel management strategy, which was the main objective to support the CNE safe operation. The main objectives of the assessment were to:

- *Determine the overall risk evolution early enough due to potential PT / CT contact which could result in the formation of hydride blisters, whom growing and cracking, could lead to the PT rupture.*
- *Evaluate the risk reduction that would result from an inspection and / or maintenance of the fuel channels.*

The key distributions of the creep ratio and end-slopes of the PTs used in the 2011 assessment were derived using a benchmark process against the gap measurements performed in the 2010 scheduled outage.

The general approach used in this evaluation was to determine the probability of blister's cracking resulting from the failure process of the PT / CT contact causing blisters. Due to the fact that the blister cracking probability is depending of the time, it was considered 232,000 equivalent full power hours (EFPH) as the evaluation period, which exceeds the EOL and provides a margin for accounting the uncertainties in the repositioning activities.

As stated in the applicable standard, CSA N285.8-05, the allowable frequency of blister cracking is 0.0025 per reactor-year. In order to fulfil this criterion, the inspection or maintenance performed in the 2011 campaign was focused in maximizing the risk reduction to zero in eleven fuel channels. So, regarding this issue, the whole reactor core is fit for service until EOL.

As result of the analysis carried out, a group of channels were identified for PT/CT inspection, with the intention to reach 232.000 EFPH. Four of them, had to be repositioned and fitness for service about contact and blister formation for the entire core was demonstrated.

3.14.2.2.2. Pressure Tube Inspection and Scrapping

CNE has undergone a total of four scrape campaigns of the PTs. The last one was in 2010 when ten PTs were visited. Measurements of the sag and the gap between PT and CT were performed in order to establish the fuel channel deformations, and consequently tune the models of the contact determination and channel deformation calculation programs. Until now these values were taken as the average from measurements carried out in other reactor. This improvement will allow estimating more accurately the PT / CT contact before the EOL.

The results showed that measured values of deuterium concentration have satisfied acceptance criteria for the hydrogen equivalent concentration level and the increase in hydrogen concentration rate. These acceptance criteria complies with the requirements of CSA N285.4-05 Clause 12.3.5.2 for PTs material surveillance and the predicted values fulfil the criteria for an evaluation period which ends at 240,000 Hot Hours.

Besides, the results from the different scrape campaigns including the 2010's campaign were the input to develop a capped Arrhenius equation with time dependence providing a reasonable representation of the relationship between deuterium concentration, time and temperature of the PTs. This model was used for engineering assessment, like the probabilistic assessment of PT/CT contact (see section 3.14.3.1.4.1.) and the susceptibility to blister formation, in order to show that fuel channels are fit for service.

During the 2011 scheduled outage, additional dimensional inspections were performed on channels with probability of PT/CT contact in order to know which channels must be repositioned. Fifteen channels were inspected for PT/CT gap, PT diameter, PT length and PT sag. Fourteen additional channels were inspected according were mentioned above, plus were inspected the existing gap between the CTs and the nozzle of the liquid injection shutdown system (LISS). Regarding this last issue, using the fuel channel specific gap measurements in the regression analysis model and the inspection results, it was predicted that four CTs will be in contact with LISS nozzle before EOL. Therefore, the ARN has required the licensee to implement mitigation actions to ensure that the LISS is consistent with the corresponding safety analysis assumptions.

During 2012 schedule outage an ISI was performed. As a result of the dimensional inspections, fitness for service for diametric expansion, thickness and elongation was demonstrated at least until 225,000 EFPH. In order to continue the operation until 232,000 EFPH some corrections due to elongations could be necessary. In regard to volumetric inspection, all flaws that have required disposition were assessed until 232,000 EFPH in accordance with CSA N285.8-10 with 13 heat-up/cool-down cycles.

3.14.2.2.3. Calandria Tube-Liquid Injection Shutdown System (LISS) nozzles contact

As a result of inspections conducted in 2010, it was observed that CTs from several fuel channels were close to the nozzles N° 2 and 5 of the SDS # 2, and in two cases the contact in a short term was predicted. The approach between the two tubes is due to differential growth that each of them suffers. To solve this situation two PTs were changed, corresponding to the fuel channels with contact prediction, thus reducing the deflection of the assembly. To ensure the CTs fitness for service, a thickness measuring was performed, which demonstrated the lack of wear and contact.

However, the prognosis of contact over time of other channels limited the operating possibility of the reactor beyond the design life. Because the plant conditions allowed assuming its operation extended beyond the design life, a series of dimensional inspections of the critical channels was scheduled for late 2011, with the intent to accurately determine the evolution of the deformation over time for each channel and thus take corrective actions and demonstrate the fitness for service. This inspections were completed by the end of 2012.

Besides, studying the possibility of contact, evaluations were conducted to determine the time required so that a possible contact CT/LISS affects the CTs' integrity.

As an alternative to the corrective action of changing the PTs, a comprehensive analysis was performed to justify the operation with some channels without fuel, thereby decreasing the deflection of such channels, due to the decreased weight of the fuel. In addition, because there is no fuel, the axial creep accelerated by neutron flux stops. In order not to alter the cooling conditions of the fuel channel, a flow restrictor was designed to produce a pressure drop similar to the fuel, which was already tested in a 1x1 mock up under simulated operating conditions.

From April 2012, the operation of the reactor was implemented with some empty channels, installing restrictors in them to correct the pressure drop. In the scheduled outage carried out at the end of 2012, inspections were completed to adjust the parameters that allow assess of the contact times and define the corrective actions necessary to continue the operation as well as to fulfil the current regulations. Until now, to avoid CT / LISS contact, three PTs were changed and four fuel channels have been emptied, where the respective restrictors were placed.

3.14.2.2.4. CNE Feeders

An important task of CANDU reactors is based on the periodic inspection established by corresponding standards with new revisions based on external operating experience. During each planned outage, several areas (e.g.: first and second bends, area next to graylock weld) of feeders are inspected, with the objective of determining the wall thickness reduction produced by flow-accelerated corrosion (FAC). Also, crack detection at feeder bends by means of ultrasonic techniques has been carried out.

The inspection is performed according to what had been established by the ISI program during scheduled outages. The inspection methods applied, and qualified personnel involved are in accordance with the procedures and guidelines from COG.

This monitoring must be done during every planned outage, in order to assure that no feeder operates with a wall thickness lower than the minimum values allowed, and to comply with the crack detection program.

During the 2011 and 2012 planned outages, assessments in the wall-thickness of the feeders according to the guidelines of ASME code and EPRI were performed.

Based on inspection data collected from past inspection campaigns, thickness predictions for all outlet feeders were carried out in 2011. The EOL thickness predictions were performed based on a "Stop-Gap approach" applied to the past measured thickness data collected from thirteen inspection campaigns. A representative sample of fifty outlet feeders was analysed considering all the design basis loading conditions. The results identified nineteen feeders that did not meet one or more of the requirements established in the ASME Boiler and Pressure Vessel Code Section III, NB-3600. The nineteen feeders were tagged as candidates for disposition by further detailed finite element analysis of the tight radius bends. Within this scope, thirteen feeders have been selected based on the worst margin of exceedance with respect to the code stress and fatigue limits. In order to address overstress condition for these thirteen feeders, two significant enhancements were conducted:

- i) Improved thickness predictions using industry-wide accepted Bayesian approach, together with detailed thickness inspection data collected from the previous outages, including data collected from the 2011 inspection campaign results.*
- ii) Improved modelling and analysis techniques to show stability of postulated cracks.*

The evaluation results determined that eleven feeders meet the ASME Section III Code stress limits, and hence are fit-for-service to the next period of operation (October 2013). By meeting the ASME Code stress limits, these feeders maintain the code safety and design margins, and as such, there are no implications to add any enhanced inspection requirements beyond those already specified via CSA N285.4 periodic and in-service inspection rules. Besides, two feeders are also fit-for-service to the next period of operation (October 2013) as they meet the rules of Fitness-for-Service Guidelines for Feeders in CANDU Reactors and there is no need for their replacement. This last statement is supported by the demonstration of the leak before break (LBB) behaviour which is being updated by the licensee.

NA-SA will continue to execute the current inspection plan in planned outages; if as a result the need for closer monitoring or corrective actions on any particular feeder arises, a specific inspection plan will be implemented for that purpose.

3.14.2.2.5. CNE Steam Generators

The CNE SGs were designed and manufactured by Babcock & Wilcox (B&W). Some of their components, mainly the Tube Support Plates (TSP) and the U-Bend supports of tube bundles, known as "scallop bars", were constructed in carbon steel.

Due to the FAC degradation mechanism, which was not well known at the moment of design and construction, and after more than 20 years of operation, the supports started to degrade producing SG's tube damage due to fretting wear in the U-Bend. This has caused several unplanned outages due to leakages in tubes, as well as a progressive degradation and loss of function of TSP.

Since 1996, damages were observed in tubes due to fretting wear with scallop bars at tube bundles U-Bend. Besides, in a study performed by B&W, this mechanism was recognized as the problem's root cause; also this study predicts future degradation due to FAC.

This encouraged CNE to take corrective and mitigation actions concerning SGs as part of the PLIM programme. In so doing, NA-SA contracted B&W, who designed auxiliary supports known as anti-vibration bars (AVB's). These components consist of three comb-shaped supports, for each SG, which have to be inserted in the U-bend of tube bundles.

When AVB's were installed in 2004's planned outage, during quality assurance boroscopic inspections performed in order to verify their correct installation, it was observed that the cold leg side of TSP were considerably damaged, and those on the hot leg side were partially obstructed. This showed that the steam/water flow was not uniform, and this accelerated FAC degradation in the cold leg side.

In addition, in order to introduce corrective actions concerning water chemistry, new chemical analysis were performed.

Considering the detected degradation in SGs, the ARN required an exhaustive assessment of the components involved, and verification of acceptance criteria established for tubes, support structures and SG internals in order to determine safe operation until the next inspection, with an acceptable safety margin.

Taking into account the regulatory requirements and findings during visual inspections in the secondary side, it was required to implement, four inspection ports between the upper TSP in each SG, in order to inspect and evaluate the TSP state, and eventually decide their cleaning.

Inspections performed from these new ports, showed sludge obstructions in the hot leg side of TSP, and variable degradation in the cold leg side of TSP (the more important degradation was observed in the higher TSP). These inspections allowed the construction of TSP degradation/obstruction maps, which were used as a reference to determine degradation rates in subsequent inspections.

Due to the degradations observed in the cold leg side of TSP, the ARN required NA-SA an assessment on structural integrity in order to verify the SGs safe operation. The assessment was given to AECL, and included evaluations performed specifically for CNE.

As a consequence of the occurrence of three unplanned outages during 2006/2007 due to tube failures in all cases localized in the U-Bend zone, the plugging criterion was changed to a more conservative position, extending the inspection zone taking into account the observed defects, going further than what was required by regulations.

During the planned 2007 outage several corrective actions were undertaken to mitigate FAC, following the recommendations of studies performed by AECL.

- Water-lancing of the hot leg side of TSP, in order to clean “broached holes” and normalize secondary side flow.
- Change of the amine used in AVT (All Volatile Treatment) treatment, in secondary circuit, in order to increase pD. Also a Cooper monitoring program was implemented, as well as a more exhaustive cation monitoring in the steam generators blow down. These corrective actions results in a reduction of the FAC rate of the cold leg due to the more alkaline pH (reduction of FAC rate by a factor of 0.5) and cleaning of broach holes (reduction of FAC rate by a factor of 0.67).
- Additional boroscopic inspections to improve the TSP degradation maps, built from the results of Eddy Current (EC) Inspections.
- EC Inspection to determine TSP degradation rate, and preventive plugging of tubes in degradation zones of TSP corresponding to levels L5 (ligament breach of two ligaments) and L6 (ligament breach of three ligaments), in agreement to AECL recommendations.
- Program to follow-up the vibration conditions and preventive plugging due to fretting in U-Bend.

Due to preliminary results obtained in visual inspections that were being performed, and considering unplanned outages due to tube failures, the ARN decided to re-assess the SG operative conditions, to perform a detailed analysis of degradation rate, to assess the structural integrity considering the SG current state and including the consequences of seismic occurrences and other events that cause sudden depressurization of the secondary circuit.

Consequently, the main tasks to be undertaken by NA-SA, related to SG safe and reliable operation, consisted of additional assessments concerning structural integrity, taking into account the new degradation maps (2007), evaluating the condition of TSP and comparing it with the result of previous inspections. This were performed in order to identify future hazard zones that could jeopardize the safe operation of components, and to allow timely corrective actions. In these studies, earthquakes and main steam line break were considered. The main conclusions of the study were:

- The condition of the SG TSP was considered acceptable, both for normal operation and for transients caused by design basis accidents.
- The result of the analysis of flow induced vibrations (FIV) for full power operation indicates that, with the TSPs condition, the SG tubes would remain stable without significant wear (excluding potential broken scallop bars), without breaks for fatigue or damage to tubes due to interaction among them. In main steam line break conditions, the initiation of fatigue cracks is not expected and it is concluded that the tubes will remain stable.
- The possibility of damage in the tubes due to weak or loose scallop bars is not excluded. However, any damage from this cause would not be significant in terms of safety. This is because this mechanism produces a small localized defect that produces a leak before the defect becomes large enough to put a tube in risk of rupture.

The following recommendations arising from AECL studies (in accordance with the SGs Life Management Program) were implemented during the CNE 2008 planned outage:

- a. All SG tubes were inspected using conventional probe (bobbin probe).
- b. Other inspections were performed with X - Probe to the tubes in the TSP critical degradation area, to tubes with relevant indications of degradation characterization, and to some tubes in the minimum radio U-Bend area.
- c. A minimum number of tubes located in the sediment accumulation area were inspected with a rotating probe.
- d. Preventive plugging of those tubes with L5 and L6 degradation levels in the TSPs zone.
- e. The moisture separators were inspected to determine their state in order to assess the feasibility of reusing them during the extended plant life.
- f. Loose parts were detected and removed from scallop bars located in the SGs U-Bend zone.
- g. The degradation progress was verified in the TSP (cold side) due to FAC, and fouling in the TSP (hot side) in all SGs, to assess the effectiveness of the "water-lancing" (performed in the 2007 scheduled outage) and the modifications in the feed-water chemistry.
- h. A dimensional topographical mapping of the connection points was performed by laser scanning, to ensure the correct and accurate installation of the new SG cartridges.
- i. A RB and service building dimensional analysis to rooms associated with the SGs replacement were performed for the subsequent planning of activities.

In order to authorize the extension of the operating license, in 2011 ARN required a report including the assessment of the state of degradation of SG's internal components, and fundamentals that permit safe operation. This report should be submitted before plant start-up after each programmed outage. Apart from that the plant must submit, within 180 days following the mentioned outage completion, a report assessing the structural integrity of the SGs' tubes as well as the support plates, considering the current state of the SG's and including the consequences of the occurrence of earthquakes and events involving sudden depressurization of the secondary.

During scheduled outages carried out in 2011 and 2012, CNE conducted inspections and assessments for internal components degradation of the SG's; the corresponding reports were presented to the AR.

NA-SA has performed the required assessment and as a result the document "Reservoir Steam Generators: Stress Analysis of Degraded Tube Support Plates and U-Bend Tubes in 2011 Condition under Seismic plus MSLB Loading" has been generate. This document recommends continuing the GV's operation subject to:

- *Conduct inspections to assess the SGs' condition in the next scheduled outages and;*
- *Based on the mentioned inspections, take necessary corrective actions, such as plugging of tubes having failure risk.*

In Addition, NA-SA decided to perform visual inspections from the GV's inspection ports to the TSPs.

As a result of SG's Life Assessment, it was concluded that SG's life extension is not feasible due to TSP FAC degradation. Steam drums will remain, while SG's cartridges and tubes will be replaced. Also, the moisture separators are not in conditions to face LTO and will have to be replaced (*see section 3.6.5.1*).

Meanwhile, several recommendations resulting from the SG's Life Assessment were implemented related to maintenance, inspections, chemical control and structural analysis, in order to achieve a safe design life.

The design changes foreseen to be implemented to the new SGs mainly respond to the following factors:

1.- Plant repowering:

- *Owing to the power increase from 2015 MW(t) to 2064 MW(t) the thermal transfer surface will be increased and;*
- *The feedwater inlet temperature and pressure increases.*

2.- Seismic re-qualification:

- *The seismic qualification strategy proposed by the plant designer was adopted and was included replacing the GV's cartridges into the Minor Modifications category.*

- *To comply with the new seismic requirement, the SGs of Qinshan NPP were adopted as reference design. This decision implies to carry out changes in both the internal supports of the SGs' tube bundle as well as in the anchorages of the external supports on the casing of the SGs.*
- *Increase of the thickness of the enveloping casing pressure near the side external supports.*

3.- 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.*
- *Installation of a new inspection port in the region of the pipes' bend.*
- *Three inspection ports for TSPs.*
- *Two new inspection ports at the preheating region.*
- *An additional inspection port for water lancing.*
- *Increased number of nozzles level measurement in wide range.*
- *Increased number of level measurement nozzles in wide range.*
- *Welded segments of the SGs' separators plates (baffles).*
- *Feedwater's chemistry.*
- *The pressure envelope components composed by carbon steel alloy (wet surfaces during service), will have a minimum of 0.2% Cr on their chemical composition.*

3.14.3. SAFETY ASSESSMENT

Safety assessments concerning operation cover all the plant operating modes and include a periodic revision of failure modes of SSCs, identifying the consequences of such failures as well. As the NPPs have been operating for a long time, and some original operation safety criteria were different from those used nowadays, it is necessary to make an additional effort in order to take into account the application of new safety criteria.

On one hand, safety assessments related to operation are mainly focused on the analysis of operational incidents and accidents that may occur in this plant or any other, the determination of the root causes and the evaluation of corrective actions in order to avoid their recurrence. On the other hand, the operation policies and principles are also analysed taking into account safety issues, the surveillance program, the maintenance program and the operating manual.

Two complementary methods are mainly applied in the evaluations: the deterministic and the probabilistic one.

The deterministic method enables the knowledge of installation and safety systems response in operational incidents taken into account as design basis. Proven engineering methods to predict the course of the events and its consequences are used for the analysis, and they comprise disciplines such as: thermo-hydraulic analysis, reactor physics, structural integrity, system control and human factor analysis.

The probabilistic method includes the evaluation of a number of conceivable accidental sequences and their radiological consequences, the reliability analyses (basically related to safety systems) and the identification of any weakness in the NPP design and operation that could contribute to risk.

The main deterministic and probabilistic assessments done to SSCs during the period corresponding to this report are shown in subsections 3.14.3.1. and 3.14.3.2.

3.14.3.1. DETERMINISTIC ASSESSMENTS

3.14.3.1.1. Analysis of loss of safety functions

As a consequence of the Fukushima accident and with the purpose of applying the corresponding lessons learned, the Regulatory Body requested performs a stress test to each Argentinean NPP (see Annex VIII) which among others safety related issues, includes:

Loss of Off Site Power (LOOP)

A total loss of external power supply for a long term, considering that the site will remain isolated for 72 hours to the possibility of heavy materials supply by any transport means and that the portable light equipment could get to the site 24 hours after the start of the event. The evaluations requested were the following:

- The current design features to face up a LOOP event, as well as the internal power supply systems designed to deal with it.*
- The time during which the above mentioned internal power supplies can operate without any external support.*
- The actions that are necessary and are planned to extend the operating time of equipment for internal power supply (e.g. filling of fuel tanks for DGs, etc.).*
- Measures foreseen to increase the plants robustness, such as system modifications, procedure modifications, organizational arrangements, and so on.*

Station Black Out (SBO)

The following two situations were considered in these analyses:

- LOOP + Loss of backup power supplies.*
- LOOP + Loss of backup power supplies + loss of any other alternative power supply.*

For each of these situations it was required:

- Provide information about measures envisaged in the design for these accident scenarios.*
- Provide information on the batteries capacity and duration. Analyse the consequences of their total loss.*
- Indicate how long the plant can withstand a SBO with no external support before the occurrence of severe damage to the fuel become inevitable.*
- Indicate external actions foreseen to prevent damage to the fuel.*
- Identify potential extreme situations which may occur, indicating the additional measures that could be incorporated to avoid their effects or to increase the robustness of the plant (modification of system or procedures, organizational arrangements, etc.).*

Loss of heat sinks

Analysis and evaluation of the sequential loss of the heat sinks designed to cool and transfer the residual heat from the reactor core and the spent fuel pool under foreseen circumstances were performed. It was required to assume that the functionality of each existing heat sink is lost and that the site will remain isolated for 72 hours to the possibility of heavy materials supply by any transport means, although it is assumed that portable light equipment could get to the site 24 hours after the start of the event.

Therefore, it was required to provide a description of the existing provisions in the design to avoid the loss of different sinks, for example, water intakes in several places, etc. For these scenarios, it was required:

- Indicate how long the plant could handle the situation without outside help before a severe damage to the fuel is inevitable.*
- Provide information on existing provisions in the design and internal actions to be performed.*
- Indicate external actions designed to prevent fuel damage.*
- Identify potential cliff edge effects which may occur, indicating the improvements that could be incorporated to avoid their effects or to increase the robustness of the plant (modification of systems or procedures, organizational arrangements, etc.).*

Loss of heat sinks coincident with SBO

The following analysis and evaluations were required:

- How long the plant can withstand a loss of all heat sinks coincident with SBO without any external support, before severe fuel damage is inevitable.*

- The existing provisions in the design and internal actions to be performed.
- Indicate external actions designed to prevent fuel damage.
- Identify potential limit situations which may occur, and when these would occur, indicating the additional measures that could be incorporated to avoid their effects or to increase the robustness of the plant (modification of systems or procedures, organizational arrangements, etc.).

Expected responses of the NPPs safety systems under conditions of loss of power supplies and heat sinks were evaluated in the framework of the stress tests carried out. Therefore, the CNA I, CNA II and CNE safety margins assuming the occurrence of a sequential loss of the defence in depth lines due to extreme initiating events occurrence were reassessed. This assessment included the long term evolution of severe accidents and the recovery capability of both the power supply and the water supply until a stable plant condition is reached. Considering the above mentioned, the following events have been analysed:

3.14.3.1.1.1. Loss of Off Site Power (LOOP)

As a result of the LOOP evaluation the Licensee has decided to implement the following improvements:

CNA I

- **Restoration of external power supply**

In case of a network collapse, the company responsible for the supply management (CAMMESA); has a procedure for CNA I supply that imposes its priority due to safety requirements of nuclear installations.

- **Passive components control**

The proper functioning of the program for the verification of the vacuum breakdown/siphons associated with cooling was checked, finding them in adequate condition. Besides, it was added to the periodic inspections program, the control of the functionality of the vacuum breakdown/siphon system associated with the pipes of the cooling systems or the inventory control of the spent fuel storage pools. Including the above mentioned issues, a procedure related with passive components control increasing the frequency of tests and inspections was implemented.

- **New Emergency Power Supply System (EPS)**

A new EPS system is under construction which is housed in a new separate building, equipped with 3 new diesel generators (DGs) with a capacity of 4,000 KVA each one, feeding to 6.6 kV bus bars. The design includes the following improvements comparing with the former EPS:

- More spatial separation among the DGs and among electrical bars (improve response to common cause failures).
- Protections, redundancies, ignition and refrigeration systems of the DGs.

At present, the assembly stage has been completed and, the auxiliary systems commissioning is on-going, being foreseen to be finished by the end of 2013. The connection to the plant is estimated to be done in 2014.

- **EPS additional modification**

Duplication of the coupling breakers between normal and secured bus bars to improve the supply of the emergency current bus bars. This improvement is foreseen to be implemented by the end of 2014.

- **Electrical interconnection between normal bars of CNA I / CNA II**

Currently an electric interconnection between CNA I and CNA II NPPs exists. It was foreseen to be disassembled during CNA II commissioning. The objective of maintaining the interconnection is being evaluated in order to both plants has another alternative external power supply. Thus the CNA I could be connected to the CNA II 500 kV lines with two redundancies. This improvement is foreseen to be implemented by the end of 2015.

- **Assessment of the external electrical supply lines availability**

The availability of the external power lines including high voltage interconnection lines 220 kV and 132 kV is foreseen to be re-assessed. This improvement is foreseen to be implemented by the end of 2015.

CNA II

- **Restoration of external power supply**

Similar to CNA I case.

- **Assessment of the external electrical supply lines availability**

The availability of the external power lines including high voltage interconnection lines 220 kV and 500 kV is being re-assessed. This improvement is foreseen to be implemented by the end of 2013.

- **Review and improvement of emergency procedures**

Review of procedures to extend the operating time of the DGs using additional fuel tanks, including the maintenance and testing programs as well as to ensure that the minimum reserves are maintained by the program of inspections and tests. This improvement will be implemented by the end of 2013.

- **Electrical interconnection between normal bars of CNA I / CNA II**

Similar to CNA I case. This improvement is foreseen to be implemented by the end of 2015.

CNE

During the plant shutdown for the refurbishment activities (2014/2016), required for the plant life extension, some improvements related with the power supply systems will be implemented. The more relevant are the following:

- **External power supply protection devices**

Protections will be improved at 500 kV station, bus bars, lines, circuit breakers failure. These improvements are foreseen to be implemented by the end of 2013.

- **Upgrade of the EPS system**

A new EPS system, including the replacement of the existing 50 kW / 75 kVA emergency DGs for new DGs with greater capacity allowing feed the pumps of the ECCS. This task is planned to be implemented by the end of 2015.

- **Class III DGs**

Replacement of the DGs belonging to the uninterruptable power supply system (Class III), including the modification of the building which houses them, is scheduled to be implemented by the end of 2015.

3.14.3.1.1.2. Station Black Out (SBO)

As a result of the SBO studies, the Licensee has decided to implement the following improvements:

CNA I

- **Alternative power sources**

A mobile diesel generator (MDG) to supply essential consumption required to face with severe accident situations caused by a SBO occurrence will be implemented as to provide alternatives to the existing sources for secured electricity supply. It also must be available a nearby and safe storehouse for the MDG and the associated accessories / devices to allow a quickly connection between the MDG and the supply lines in order to ensure the cooling of the core and the spent fuel storage pools in the long term. This improvement will be implemented by the end of 2014.

- **Preventive strategies to avoid core damage**

The following strategies to prevent core damage associated with the loss of cooling function have been undertaken:

- *Coolant inventory reposition by the pressure and inventory control system (TA) in small LOCA conditions (from a mobile diesel generator -MDG- supply, under analysis).*
- *Water supply to the SGs through the SHS system (SSC/RX) in different accidental scenarios. It is in the detailed engineering stage.*

- Establish a core cooling mechanism, preventing that air from the boron injection system (TB) entry to the PHTS in case of a SBO scenario. This improvement is being reviewed and is foreseen to be implemented by the end of 2014.
- Strategy to be applied to face a 220 VDC power supply failure was already implemented.
- Strategy to be applied in case of voltage drop in the 24 VDC lines was already implemented.

- **Fuel elements integrity assessment**

The SBO impact on the fuels that could be housed inside the refuelling machine considering its full load is being assessed. The study is foreseen to be implemented by the end of 2014.

The measures necessary to ensure the integrity of the fuel housed inside the refuelling machine must be implemented in case the assessment results mentioned in the previous bullet will indicate that the fuel could be damaged. This improvement is foreseen to be implemented by the end of 2015.

- **Extension of the batteries availability**

A reduction of the loads that are feed by batteries to extend the availability of the batteries until it is provided an alternative energy supply will be implemented by the end of 2014.

- **Instrumentation & Control (I&C)**

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.

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 BDBA. This improvement is still under analysis.

CNA II

- **Alternative power sources**

A MDG to supply essential consumption required to face with severe accident situations caused by a SBO occurrence will be implemented as to provide alternatives to the existing sources for secured electricity supply. It also must be available a nearby and safe storehouse for the MDG and the associated accessories / devices to allow a quickly connection between the MDG and the supply lines in order to ensure the cooling of the reactor core and the spent fuel storage pools in the long term. In particular, the MDG should be able of be connected to a breaker located in the emergency supply bar, including a facility for the connection to each emergency output. This improvement will be implemented by the end of 2014.

- **Fuel elements integrity assessment**

Similar to the CNA I case.

- **Connection among the DGs of CNA I and CNA II**

The necessary arrangements to allow that by mean of manual actions, the DGs of CNA I provide power to CNA II through two bars of the 6.6 kV normal network and the four bars of the 6.6 kV emergency networks (implement the connection of one of the new DGs of CNA I to CNA II). This improvement is still under analysis.

- **Extension of the batteries availability**

Improvements to extend the electrical supply from the batteries after an SBO occurrence have been foreseen to be implemented by the end of 2015.

CNE

During the plant shutdown for the refurbishment activities (2014/2016), required for the plant life extension, some improvements related with the SBO will be implemented. The more relevant are the following:

- **Abnormal event procedure**

An abnormal event procedure to respond to the loss of cooling and / or loss of inventory from the spent fuel storage pool was implemented. This procedure includes the actions required to monitor the coolant level and the pool temperature from the secondary control room supposing that the

main control room and the pool building are inaccessible. Besides, to face a loss of cooling or inventory in the long term, suitable actions have been included, to replenish water from alternative sources. These improvements are foreseen to be implemented by the end of 2013.

- **Fuel elements integrity assessment**

The SBO impact on the fuels that could be housed inside the refuelling machine considering its full load is being assessed. This study is foreseen to be implemented by the end of 2014.

The measures necessary to ensure the integrity of the fuel housed inside the refuelling machine must be implemented in case the assessment results mentioned before will indicate that the fuel will be damaged. This improvement is foreseen to be implemented by the end of 2015.

- **Alternative power sources**

A MDG to supply essential consumption required to face with severe accident situations caused by a SBO occurrence will be implemented as to provide alternatives to the existing sources for secured electricity supply. It also must be available a nearby and safe storehouse for the MDG and the associated accessories / devices to allow a quickly connection between the MDG and the supply lines in order to ensure the cooling of the core and the spent fuel storage pools in the long term. This improvement will be implemented by the end of 2015.

- **Extension of the batteries availability**

Improvements to extend the electrical supply from the batteries after an SBO occurrence have been foreseen to be implemented by the end of 2015.

3.14.3.1.1.3. Loss of heat sinks

As a result of the assessment above mentioned, the Licensee will implement the following improvements:

CNA I

- **Alternative water sources**

Provide alternatives water sources to the existing ones for reserve secured water supply (such as reservoirs, pools, tanks, etc.) to face severe accident situations caused by the loss of heat sinks. It also must have the appropriate accessories/devices to connect these water sources with the corresponding pumps and supply lines in order to ensure the cooling of the core and the spent fuel storage pools in the long term by implementing the following:

- The process water refilling of the spent fuel storage pools by using an alternative reservoir, such as groundwater, existing tanks or other sources. Besides, this process will include the monitoring of the water level and temperature of the pools. This process will be implemented by the end of 2013.
- Install an independent pump assigned to the process mentioned before that allows extracting groundwater to fill the spent fuel storage pools. The pump electrical supply will be possible from a secured bar and from a MDG operated by means of a manual simple connection. Furthermore, it is foreseen to be implemented an alternative power supply system to feed the necessary I&C to monitor the relevant parameters of the spent fuel storage pools from outside the pools building. This improvement will be implemented by the end of 2015.
- Replenishment of water inventory to the SGs through the SHS system in case of simultaneous loss of the feedwater tank, the residual heat removal chain and the water injection secured system to the SGs. Also, it should provide replenishment of water inventory to the SHS tank in those cases where the SHS integrity is not affected. The replenishment of water inventory for the above cases must be implemented by injecting water from the groundwater using one of the pumps belonging to the water supply system, foreseeing that the involved components must be fed through a MDG in case of SBO is coincident with the unavailability of the DGs belonging to the SHS system. This improvement will be implemented by the end of 2014.
- Replenishment of water to the SHS system to ensure its operation for 72 hours without be necessary perform external actions. This improvement will be implemented by the end of 2015.

CNA II

- **Alternative water sources**

Provide alternatives water sources to the existing ones for reserve secured water supply (such as reservoirs, pools, tanks, etc.) to face severe accident situations caused by the loss of heat sinks. It also must have the appropriate accessories/devices to connect these water sources with the corresponding pumps and supply lines in order to ensure the cooling of the core and the spent fuel storage pools in the long term by implementing the following:

- *Provide an alternative water reservoir that allows maintaining the water supply to heat removal through the SGs and cooling the spent fuel storage pool in the long term.*
- *Additional system to water replenishment of the spent fuel storage pools from an alternative reservoir such as the groundwater, existing tanks or other sources.*

These improvements are foreseen to be implemented by the end of 2015.

CNE

During the plant shutdown for the refurbishment activities (2014/2016), required for the plant life extension, some improvements related with the loss of heat sinks will be implemented. The more relevant are the following:

- **Alternative water sources**

Provide alternatives water sources to the existing ones for reserve secured water supply (such as reservoirs, pools, tanks, etc.) to face severe accident situations caused by the loss of heat sinks. It also must have the appropriate accessories/devices to connect these water sources with the corresponding pumps and supply lines in order to ensure the cooling of the core and the spent fuel storage pools in the long term by implementing the following:

- *The water replenishment of the spent fuel storage pool through a connection from outside the building pool that includes an isolation valve and hose coupling from the fire fighting system was implemented.*
- *Installation of a facility to connect a fire truck from outside the building pool to replenish water in the spent fuel storage pool to face events of loss of cooling, flow loss or SBO was implemented. This improvement is foreseen to be implemented by the end of 2014.*
- *Provide two mobile cisterns containing 25,000 litres of stored water each. This improvement is foreseen to be implemented by the end of 2013.*
- *Provide a water supply line to the calandria vault from outside the reactor building. This improvement is foreseen to be implemented by the end of 2015.*
- *The installation of a connection through a hose line from a fire truck to the ECCS pipes to allow the water addition to the dousing tank for replenishment the feedwater to the SGs' to allow its cooling for at least 72 hours. This improvement is foreseen to be implemented by the end of 2015.*
- *Provide an additional fire truck containing 17,000 litres of water to be used for replenishment. This improvement is foreseen to be implemented by the end of 2015.*

- **Modifications to the Emergency Water Supply (EWS) system**

The EWS system will be modified by replacing the existing pumps and the corresponding diesel engines by two new pumps with higher capacity driven by the new DGs belonging to EPS system, so with the capacity to allow feeding the ECCS heat exchanger (low pressure). This modification will include the replacement of pipes and the valves that connect the EWS pump house with the service building as well as the duplication of the valves that supply emergency water to the core and the SGs. This improvement is foreseen to be implemented by the end of 2015.

- **Abnormal event procedure**

Operational procedure for abnormal events aimed to increase up to seven days the water replenishment availability to both the dousing system and the SGs, by using the EWS pumps. This improvement is foreseen to be implemented by the end of 2015.

3.14.3.1.1.4. Loss of heat sinks coincident with SBO

It was verified that the improvements proposed by the Licensee showed in the previous items for the cases of loss of offsite power; SBO and loss of heat sinks covers the loss of heat sinks coincident with SBO scenarios for CNA I, CNA II and CNE, excluding the following proposal corresponding to CNE:

- **CNE**

During the plant shutdown for the refurbishment activities (2014/2016), required for the plant life extension, some improvements related with the loss of heat sink will be implemented. The most relevant are the following:

- **Instrumentation & Control (I&C)**

- *Install a I&C system in the secondary control room to allow monitoring the level and water temperature of the spent fuel storage pool, independent from the I&C system existent in the main control room, so that measurements could be repeated in both control rooms. This improvement will be implemented by the end of 2014.*
- *Improvements in the I&C used to measure the following parameters to be implemented by the end of 2015:*
 - *Hydrogen flammability.*
 - *Sub cooling margin of the PHTS.*
 - *Moderator level.*
 - *Water level in the calandria vault.*
 - *Pressure inside the containment.*
 - *Radiation level.*
 - *Hydrogen flammability inside the containment.*
 - *Water level in the reactor building sump.*

3.14.3.1.1.5. Regulatory Actions

The proposed improvements to cope with design weaknesses were evaluated. As result of the review, the Regulatory Body concluded that the licensee of CNA I; CNA II and CNE have performed satisfactorily the required stress tests, considering that:

- *No significant weaknesses have been identified that require taking urgent actions.*
- *For the purpose of increasing the capacity to respond against loss of safety functions, the Licensee proposes to implement a set of improvements as well as the corresponding implementation schedule that are acceptable.*

Besides, the Regulatory Body issued a regulatory requirement asking improvements and modifications referred to loss of safety functions, to be implemented in each NPP. This regulatory requirement includes short, medium and long term actions; the latter reaching the year 2016.

3.14.3.1.2. Accident Management and Severe Accident Management Program

As was mentioned in previous CNS Reports, the ARN required the Licensee to develop a Severe Accident Management Program (SAMP) for both plants in operation. The initial activities were (see 2004 CNS Report):

- 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 (see 2007 CNS Report), as follows:

- Plant damage states characterization including deterministic studies related to new scenarios (or extending the existing ones) with RELAP code up to core uncovered initiation.

- Strategies proposal (preventive strategies for the time).
 - Small LOCA plus failure of low pressure emergency water injection.
 - Small LOCA plus moderator pumps failure which delays low pressure emergency water injection due to depressurization delay.
 - Blackout (SBO).
 - Unavailability of the second heat sink pumps or lack of sufficient inventory for different accidental scenarios.
 - Analysis of potential strategies related to both the use of an alternative control room in case the main control room becomes uninhabitable and the possibility of venting the containment.
- Progress in the development of the accident progression model. The experience of CNA II PSA Level 2 was transferred back to CNA I SAMP.
- In the same way, the experience gained for CNA II with Passive Autocatalytic Recombiners (PARs) was used in CNA I applications.

A schedule to continue CNA I SAMP development was approved by ARN in 2009. The CNE SAMP development has been considered integrating the tasks related to the life extension project (see 2010 CNS Report). 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 a consequence of the Fukushima accident and with the purpose of apply the corresponding lessons learned, ARN requested perform a stress test to each Argentinean NPP (See Annex VIII) 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 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 alternating 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 were showed weaknesses which make necessary to take urgent actions.

The improvements and modifications proposed by the NPPs Licensees included an implementation schedule composed by short, medium and long term actions; the latter reaching the year 2016 which were considered acceptable by the Regulatory Body. The improvements as the stress tests results include the following:

3.14.3.1.2.1. CNA I

Plant Model Using MELCOR Code

As CNA I has a unique design, there were neither applicable procedures nor accident management guidelines obtainable from similar plants. Therefore, it was decided to develop a model specific for CNA I by using the MELCOR code to simulate accidents progression including containment fail or bypass after a core meltdown and the evaluation of proposed mitigation strategies effectiveness.

Besides, a preliminary simulation model of the CNA I in-vessel stage has been developed in order to study different accident scenarios. The model was compared with CNA II results, given the similarities between both plants. The specific CNA I model is under development and is scheduled for completion by 2015.

Severe Accident Management Guidelines (SAMG)

The Severe Accident Management Guidelines (SAMG) is being re-evaluated.

WANO SOER 2011-2 recommendations

The availability of the existing SSCs for different accident scenarios was checked to meet the recommendations arising from the WANO SOER 2011-2. The Licensee compiled a list of 253 components that needed to be verified to deal with events included in the design basis, which were reviewed in walkdown and inspections performed during 2011.

Procedure for operation in perturbations and accidents

The procedure for "Operation in Perturbations and Accidents" was modified in order to include the control of critical parameters of the spent fuel storage pools.

Passive Auto-catalytic Recombiners (PARs)

The assessments to determine the specifications, location and quantity, detail engineering and required procedures of the PARs were performed. During May 2013 programmed outage 28 of 32 PARs were installed, and it is foreseen to install the rest during 2013.

Dose rate remote measurement system

A system for dose rate remote measurement, including all directions in a circle of 10 km radius connected on line to both the internal centre of emergency management (CICE) and the external centre of emergency management (CECE) was already implemented.

Strategies to reduce the containment pressure during severe accidents

The implementation of these strategies will be defined after the results of the accident progression model will be available. The strategies consider the containment pressure reduction from:

- Outside the containment by means of the annular ground ventilation.
- Inside the containment by recirculating air using its own ventilation systems.

Cooling of the RPV external side

The assessment of the RPV external side cooling considering conditions where the core is severely damaged as well as the eventual design changes necessary to supply cooling water, are foreseen to be finished by the end of 2015.

Review and improvement of accident management procedures

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

- SBO: Manual action to inject the SHS in a short period of time with a cooling ramp of 100°C/h and manually deactivate of the TB.
- 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 (UA10 D20/D21) and replace water in those pools with groundwater using one of the pumps of the potable water supply (UJ).
- Low river level. This will allow systematic manoeuvres to carry a plant outage.

These improvements are foreseen to be implemented by the end of 2014.

Severe accident management program

An emergency operating procedure for responding to an event of loss of cooling or water inventory of the spent fuel storage pools intended to monitor water level and the temperature of the pools during an emergency, as well as the possibility of recover its water inventory even in scenarios of loss of main control room, SBO, earthquakes and flooding or low water level. This improvement is foreseen to be implemented by the end of 2013.

Complete the severe accident management program including the corresponding guidelines for prevention and mitigation considering the lessons learned from Fukushima. This implies to include the strategies to face the occurrence of extreme external events beyond the design basis which leads to a loss of safety functions and to conditions of severe accident. This improvement is foreseen to be implemented by the end of 2015.

Venting filtered containment system

Demonstrate through a suitable assessment that, for any of the extreme events considered, 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. This assessment is foreseen by the end of 2014. It will be based on CNA II results from PSA L2 and similarities among others NPPs.

Install a venting filtered in the containment 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. This improvement is foreseen to be implemented by the end of 2016.

Instrumentation and Control

The I&C of the reactor core and the containment necessary to dispose of the required information for the severe accident management will be installed by the end of 2015.

Alternative cooling mode of the DGs

The present DGs are cooled by river water cooling ensured system (UK). DGs of the new EPS are air cooled.

Disconnection of electrical loads

The procedure of disconnection of electrical loads to increase the batteries duration in the actual plant condition has to be reconsidered once the new EPS be installed.

The new EPS is designed with a new concept of uninterruptible system. This concept consists in loads connected to short time interruptible bus bars and other loads connected to uninterruptible bus bars. The modification consists in feeding only necessary loads to these uninterruptible buses in order to increase batteries life.

3.14.3.1.2.2. CNA II

Cooling the RPV external side

- *RPV cooling from the external side in case core significant damage had occurred: this strategy and its effectiveness as well as the corresponding guides for severe accident management are currently under development and analysis. This task will be finished in 2014.*
- *The RPV cavity flooding with water replenishing from an alternative water supply source is foreseen to be implemented by the end of 2014.*

PARs installation

PARs installation required for the hydrogen management inside the containment is foreseen to be implemented by the end of 2013.

Venting filtered containment system

Similar to CNA I case (see section 3.14.3.1.2.1.). The assessment to demonstrate that the maximum pressure that can withstand the containment would not be exceeded and that it can continue to fulfil its safety functions under severe accident conditions will be finished by the end of 2013. In case to be necessary to install a venting filtered in the containment system, it is foreseen to be carried out by the end of 2015.

Increasing DGs autonomy

Use the fuel of the auxiliary boiler to increase the DGs autonomy. This improvement is foreseen to be implemented by the end 2014.

Switchgear building ventilation

The ventilation closure of the switchgear building (UBA) as well as the mobile purification equipment use has been foreseen. This improvement is expected to be implemented by the end 2015.

Alternative cooling mode of the DGs

The current cooling system of the two existing DGs is foreseen to be adapted by using cooling towers with air circulating with capacity up to 5 MW. This improvement will be implemented by 2015.

Disconnection of electrical loads

Strategies to disconnect unnecessary electrical loads to increase the batteries life will be implemented by 2015.

Instrumentation and Control

The I&C of the reactor core and the containment necessary to dispose of the required information for the severe accident management will be installed by the end of 2015.

Severe accident management program

Complete the severe accident management program including the corresponding guidelines for prevention and mitigation considering the lessons learned from Fukushima. This implies to include the strategies to face the occurrence of extreme external events beyond the design basis which leads to a loss of safety functions and to conditions of severe accident. This improvement is foreseen to be implemented by the end of 2015.

3.14.3.1.2.3. CNE

Severe Accident Management Guidelines (SAMG)

The Severe Accident Management Guidelines (SAMG) is being re-evaluated. On 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.

Procedure to passive components control

This procedure was already incorporated to the Operations Manual to make possible for example to verify during each operational shift the vacuum breakdown / siphons piping associated with the spent fuel pool cooling as well as to increase the frequency of the corresponding tests and inspections.

Safety system trip parameters

Improvement in the safety system trip parameters coverage is foreseen to be implemented during plant life extension refurbishment:

- Propose new additional trips to the safety systems as well as the modification of some of the existing trips which are required to improve the defence in depth against the accidents already covered by the current trips. These proposals are foreseen to be identified by the end of 2013.
- Implement the trips and modifications above mentioned. These improvements are foreseen to be implemented by the end of 2015.

PARs installation

PARs installation required for the hydrogen management inside the containment is foreseen to be implemented during the life extension refurbishment.

Venting filtered containment system

A venting filtered in the containment system is foreseen to be installed by the end of 2015.

ECCS reliability

Improvements in the ECCS reliability consisting in changes to:

- Guarantee the injection.
- Increase the system reliability operation.
- Avoid coolant leaks to the ECCS (containment by-pass).

These improvements are foreseen to be implemented during plant life extension refurbishment.

Operating Procedure for Abnormal Events

A new Operating Procedure for Abnormal Events that covers response to loss of cooling in the spent fuel pool and / or loss of inventory will be developed. This procedure shall include actions to verify the coolant level and temperature of the pool from the secondary control room in the event that the main control room and the pool room are unavailable. It shall include actions to replenish water from alternative systems (e.g. fire hydrant or fire engine) in the event of sustained loss of cooling or loss of inventory. Implementation is scheduled by the end of 2014.

Facility to connect a fire-truck from outside the spent fuels storage pool building

It will be installed a facility to connect a fire-truck from outside the pool building, which will replenish water to the pools in the events of loss of cooling, circulation or SBO. Implementation is scheduled by end of 2015.

Instrumentation and Control

The I&C located in the reactor core and the containment necessary to dispose of the required information for the severe accident management will be installed during plant life extension refurbishment.

3.14.3.1.2.4. Regulatory actions

As 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 improvements and modifications referred to severe accident management and recovery, to be implemented in each NPP (some have already been implemented). This regulatory requirement includes short, medium and long term actions; the latter reaching the year 2016.

3.14.3.1.2.5. Plant damage states characterization

In the case of CNA I, new deterministic studies using RELAP code allowed to complete the characterization and grouping of accidental sequences.

In the CNA II's PSA framework, during the 2010-2013 period, severe accidents several analyses were performed. As a result of the PSA L1, a number of relevant Core Damage States (CDS) have been identified. These CDS has been used in the calculations of the PSA L2 carried out through the MELCOR code in order to assess the severe accident progression as well as the possible radionuclide releases into the environment. The results have been used directly by the PSA L3 study.

The L1 PSA for CNE identifies a large number of accidental sequences that lead to core damage (PFS, plant final states). A total of 21 categories of PFSs have been identified. Based on past AECL experience with L2 PSAs, the grouped PFS sequences for CNE internal events L1 PSA were grouped into four general event categories or scenarios for which the accident progression will be analysed using MAAP4-CANDU. The four categories of events are:

- *In-Core LOCA.*
- *Out-of-Core LOCA.*
- *Station Blackout.*
- *Containment bypass (e.g., SG's tube rupture).*

3.14.3.1.2.6. Strategies proposal

Progress in the implementation of the proposals strategies varies from case to case. In some cases, the plant instructions have been approved and they have been included in the operators' training plan.

3.14.3.1.2.6.1. CNA I

As were reported in previous CNS reports, the strategies initially proposed were the following:

- Small LOCA strategy, the basic engineering design changes needed to implement the primary system inventory refilling strategy by means of the volume and pressurizing control system (TA) was completed.
- The engineering tasks in order to implement the recovery of the second heat sink system (SHS) capability by injecting demineralized water to SHS make up tank is being analysed.
- Blackout: This scenario was at first characterized considering excessively conservative values for the batteries discharge time. This consideration imposed a penalty in the recuperative actions model. A new version for this characterization was issued interpreting in a more realistic way the battery efficiency curve provided by the manufacturer. The instruction for the blackout scenario is under development.
- The evaluation of the strategy related to the use of an alternative control room in case the main control room becomes uninhabitable is under study by the plant engineering group. A report was issued regarding the potential strategies of the containment management during a severe accident. Different containment venting strategies were examined and the possibility of the steel containment refrigeration by internal and external circulation was considered. The potential strategy is under assessment.
- A strategy proposal for the scenario of loss of the secondary coolant system (UK) combined with an operators failure to prevent a LOCA through the use of the pressurized valve, is being analysed aimed at the recovery of the primary system inventory.

In the 2010 - 2013 period, the following preventive strategies have been undertaken to date:

Accident management measures currently available to protect the core at various stages of a scenario of loss of cooling function of the core.

In addition to the preventive strategies aforementioned, two new scenarios were identifying during execution of the review PSA Rev. 3 Level 1. These scenarios, lead to loss of current, resulting from the loss of the rectifier 220 V and 24 V which is a situation that exceeds the design basis; for this reason, subsequent handling of the plant are considered included in the PGAS preventive phase.

In the frame of the response to the regulatory requirement (RQ-38) to the Licensee of CNA I that strategy are considered as measures before the onset of damage to the fuel.

The corresponding instructions are currently in the reviewing process by the ARN.

In relation with measures after the occurrence of the fuel damage, it is possible to mention that due to the specific model for CNA I, simulated with the MELCOR code was under development, for the SBO sequence a qualitative comparison of the evolution of the phenomenology of core degradation for CNA I, was performed based on CNA II results.

Because the cooling channels and fuel elements of CNA I and CNA II are geometrically similar, and also the average power per channel and per bar, as well as the flow passage area per channel are similar, it can be concluded that the thermo-hydraulic parameters for a representative average channel are similar.

Therefore, since the initial operating conditions are also similar, it is expected that the thermo-hydraulic evolution within the pressure vessel in an accident sequence such as a SBO, be similar for both CNA I and CNA II.

For the same sequence SBO there is a comparison between the behavior of CNA II (PHWR) and a German KWU reactor of similar power. The conclusion of this study is that the calculated time to failure of the RPV for CNA II is five times greater than that of the German plant.

In the case of CNA I, the cooling of the RPV external side evaluation considering conditions where the core is severely damaged as well as the eventual design changes necessary to supply cooling water, were foreseen to be implemented by the end of 2015.

In the case of CNA II, the RPV cooling from the external side in case of core significant damage had occurred as well as the corresponding guides for severe accident management are being developed. This improvement will be implemented by the end of 2014.

In relation with the strategies for inventory replenishment, was developed only one procedure for the operator included in the Operation Manual. On the other hand were developed the corresponding SAMG D3 y DS3 and the independent review on the operator side was completed.

For the strategy to supply water to the steam generator by the RX system were developed the corresponding SAMG D1 y DS1. The stage of the independent review on the operator side was completed.

Accident management measures and plant design features for the protection of the integrity of the containment function after the occurrence of fuel damage

In this case the following management measures have been taken:

- *RPV Protection by cooling the RPV external side (see section 3.14.3.1.2.1.).*
- *Prevention of deflagration and detonation of hydrogen (see section 3.14.3.1.2.1.).*
- *Preventing overpressure inside the containment (see section 3.14.3.1.2.1.).*
- *Prevention of re-criticality:*

Criticality evaluation studies were conducted; whose main results demonstrated that, the case of re-criticality in the event of re-flooding of the core is not relevant for this type of reactor.

- *Prevention of meltdown passing through the slab:*

Regarding the prevention of the slab penetration, as mentioned previously, once the results for CNA II become available, an equivalent strategy for cooling the RPV from the outside will be analyzed, as a way to protect its integrity and stop the accident progression.

- *Alternating and direct current supply, as well as compressed air supply to the equipment needed to protect the containment integrity:*

The signal that produces the containment closure depends on the assured supply of direct current and is activated by the loss of 24 VCC. The mentioned signal sends the closure order to both valves, the pneumatic and the motorized one, so that both will close in a normal situation,

ensuring the containment insulation. In case of loss of AC, the motorized valves will not have electrical power and they will not close, causing the insulation of the containment to be produced only through the pneumatic valves. Besides, the main hatch is electrohydraulic and in case of power failure can be maneuvered manually.

Measures for accident management to deal with loss of cooling in the fuel storage pools

An emergency operating procedure for responding to an event of loss of cooling or water inventory of the spent fuel storage pools intended to monitor water level and the temperature of the pools during an emergency, as well as the possibility of recover its water inventory even in scenarios of loss of main control room, SBO, earthquakes and flooding or low water level. This improvement is foreseen to be implemented by the end of 2013.

Potential effects on other nearby plants

The potential effects of CNA I over CNA II and vice versa are being analyzed. Because CNA II is at the beginning of the commissioning, the results of these analyses are foreseen before starting the power rise phase.

3.14.3.1.2.6.2. CNA II

In the 2010 - 2013 period, the following preventive strategies have been undertaken to date:

Accident management measures and design features to protect the integrity of the confinement function after the occurrence of damage to the fuel

- The analysis of both the severe accident and the source term were used to support both the decision of the PARs installation and the corresponding design concepts (see section 3.14.3.1.2.2.).
- External Cooling of the RPV (see section 3.14.3.1.2.2.).
- Accident management measures currently available to deal with the successive stages of a scenario of cooling function loss of the fuel storage pools: It is extremely unlikely that fuel element degradation be reached, because there will be 188.5 hours to take actions to recover the cooling system. Taking into account that the worst case scenario (pool filled with all the fuels corresponding to a reactor core) is very unlikely to occur, if the spent fuel storage pools do not have the possibility of some cooling as in the case of an SBO, given the large volume of water available and its associated thermal inertia, it is estimated that there is sufficient time to recover their cooling system.
- Venting filtered containment system (see section 3.14.3.1.2.2.).
- Increasing DGs autonomy (see section 3.14.3.1.2.2.).
- Instrumentation and Control (see section 3.14.3.1.2.2.).
- Alternative cooling mode of the DGs (see section 3.14.3.1.2.2.).
- Disconnection of electrical loads (see section 3.14.3.1.2.2.).

Planned improvements

Based on the evaluations, the following planned improvements have been identified:

Additional power supply

The major improvements planned regarding the additional power supply, are the following:

- Review of procedures for extending the DGs use (see section 3.14.3.1.1.1.).
- The possibility of connecting one of the three new CNA I DGs (see section 3.14.3.1.1.1.).
- Analysis of the availability of external power supply lines including the interconnection with the 220 kV and 500 kV high voltage lines (see section 3.14.3.1.1.1.).
- Connection to mobile diesel generator (MDG) (see section 3.14.3.1.1.2.).

Heat removal through the SGs

(see section 3.14.3.1.1.2.).

3.14.3.1.2.6.3. CNE

In the 2010 - 2013 period, the following preventive strategies have been undertaken to date:

Accident management measures currently available for the reactor core protection at different stages of a scenario of a loss of core cooling function.

SAMGs correspond to mitigation, and the input conditions are such that fuel damage has already occurred or is imminent. The output condition of each guide corresponds to having reached a steady state which is specified in the same directory. This steady state corresponds to the confinement of radioactive material, which preserves at least the last barrier and the long term heat removal.

As for the mitigation strategy, considered in the SAMGs' development to stop the progression of the accident by keeping the cooling of the calandria, it consists on filling the calandria vessel or injecting water into the calandria vault and keeping it all the time flooded to maintain its integrity.

However, in the next protection level, the external phenomena of the calandria vault have been considered and the design provisions have been reevaluated, as well as the use of SAMGs to protect the containment functions have been considered.

Accident management measures and design features of the plant to protect the integrity of the confinement function after the occurrence of fuel damage.

The containment building provides a barrier containment of radioactive emissions to the environment and to the public, which is critical in the unlikely event of a severe accident. The function of protecting the containment requires the limitation of the inner temperature and pressure of the containment.

The main challenges identified are due to hydrogen production; slow the over-pressurization of the containment and interaction of the molten core with concrete of the calandria enclosure (MCCI).

One of the hydrogen management strategies consists in dilute or remove hydrogen from containment atmosphere to prevent its combustion and thus maintain its long term integrity. The hydrogen dilution with air is possible by using the compressed air tanks existing in the reactor enclosure (the set outlet pressure of the compressors must be raised to overcome the opening pressure of the tanks' safety valves) to inject air and thus reduce the concentration of hydrogen in air inside the containment.

Another of the mentioned strategies is based on the homogenization of the gases in the containment atmosphere, considering that the containment designs favor the natural circulation and, if the local air coolers (LAC) are available, through the corresponding forced air circulation.

A specific guide has been developed based on pressurizing the containment to dilute the hydrogen concentration. The various alternatives favor the increase of steam concentration within the containment.

In addition, as a second line of defense PARs will be installed to increase safety margins and ensure control of the hydrogen concentration (see section 3.14.3.1.2.3.).

The existing design features that protect the containment integrity against over-pressurization are the large containment volume (48,000 m³ volume net) and passive condensation in the structures existing in the containment building, LACs and as a result of the dousing system action. A specific SAMG is being developed to address this phenomenology and, the installation of a filtered vent system of the containment atmosphere will be carried out during the refurbishment activities foreseen to extend the plant life (see section 3.14.3.1.2.3.).

Generic assessments show that, if the core part that passed the moderator vessel boundaries is not submerged in water and cooled, the MCCI would occur at least two days after the accident initiation and, the consequent concrete penetration would occur after two days (four days from the accident occurrence). This scenario would provide enough time to implement mitigation actions to bring the plant to a controlled and stable state (the above was taken into account in the SAMGs development).

Accident management measures currently available to handle the successive stages of a loss of cooling scenario in the spent fuel storage pools.

The sustained loss of cooling in the spent fuel pool constitutes an unlikely emergency situation, which would be usually prompted by a common cause failure, such as an earthquake causing a SBO. In this scenario it would be necessary to replenish water to the pool in order to prevent discovering the fuels and avoid the potential hydrogen generation.

According to the calculations performed, it was estimated that there would be enough time to set a water supply of 1kg/s so as to keep the fuels submerged. There is no chance of re-criticality in the pool of CANDU reactors, whether the fuels are submerged or exposed.

Therefore, a loss of cooling the spent fuel pool should be successfully handled by using the Operating Procedure for Abnormal Events, which is currently being revised (see section 3.14.3.1.2.3).

3.14.3.1.2.7. Progress in the development of the accident progression model

The continuation of this task is conditioned to the progress in the CNA II Level 2 PSA modelling development tasks.

During the period from 2010 to 2013 the following activities were developed:

CNA I

According was said before in relation with measures after the fuel damage occurrence, a qualitative comparison of the phenomenology evolution of the core degradation for CNA I, was performed based on CNA II results. This allows to estimate the characteristic times related to the in- vessel progression during severe accidents. A preliminary simulation model of the CNA I containment has been developed.

A study on the CNA I containment model stability was performed, which concluded that all the pressures and temperatures are stable and the flows are stabilize in few seconds providing more cooling to structures of the containment building. The specific CNA I model is under development (see section 3.14.3.1.2.1.).

CNA II

The PSA of CNA II relies on deterministic computer simulations using the integral code MELCOR 1.8.6. for severe accident analyses. A large effort has been devoted on the L2 PSA to produce a set of MELCOR 1.8.6 version YV revision 3165, which runs according to the state of the art on a best-estimate basis starting with initial and boundary conditions as close as possible to real plant specific data and CDS definitions.

In the framework of the licensing process, and as part of the L2 PSA, the last version of the progression model with MELCOR code is currently under review by ARN.

CNE

The severe accident progression performance analysis will be performed with the approved version of MAAP4-CANDU code. The code is the Industry Standard Toolset (IST) for the accident progression analysis of the CANDU's L2 PSA and it will be the main tool for the accident progression analysis. To ensure the analyses quality performed using MAAP4-CANDU, a validation plan by the industry has been prepared in conformance with Canadian standard CSA N286.7.

3.14.3.1.2.8. Other issues related with severe accidents

In the CNA II case, the Licensee assesses the issue of the hydrogen behaviour and its associated mitigation systems (PARs) with the AREVA assistance. In order to optimize the PAR location and to assess its efficiency to face different scenarios, both the MELCOR code and plant walk-down results were used. The following three phases are being considered:

Phase I: PAR System conceptual design (finished).

Phase II: Detailed design (*finished*).

Phase III: Hardware supply (*finished*).

The experience gained *in CNA II was successfully* used as a basis for the *corresponding evaluations, to define its location and for assemble the PARs in CNA I (see section 3.14.3.1.2.1.).*

3.14.3.1.2.9. Development of Instructions and Guidelines

A new version of the methodology used in CNA I to convert the strategies into instructions and guidelines was issued. The preventive strategies from SAMP *are being* included into the Operation Manual as extensions of abnormal operating procedures (indicating interface), or as new Instructions for SAM preventive domain (Instructions P). Mitigation strategies will be included as Guidelines for

SAM mitigation domain (Guidelines M). Most of the strategies corresponding to the preventive domain will be applied by control room staff before the Internal Committee of Emergency Control team becomes available.

During the reported period, the CNA I operator developed the SAMG Instructions D corresponding to the modification of the mandatory documentation of the Operation Manual. The instructions D are divided in instruction D#, which will be used by the CICE (Internal Centre of Emergency Control) and the instruction DS#, which will be used in the main control room. Currently, the SAMG are being re-evaluated on the operator side.

Two scenarios related to the loss of direct current (DC) due to rectifier failure were identified during the last L1 PSA revision. *The instruction for the loss of a 220 V rectifier as well as the instruction for the loss of a 24 V rectifier was already issued.*

The CNE SAMP is based on the specific COG guides, developed for CANDU reactors operating in Canada. AECL provided assistance to the Licensee for developing the CNE specific guidelines, taking into account the experience of this organization in the development of SAMP Guidelines for Point Lepreau NPP. The Point Lepreau guidelines applicability for CNA I *was* evaluated by AECL with the aim to set the scope for each task. The program consists on:

- The understanding of the technical bases for the management of severe accidents for CANDU reactors.
- The study of the severe accidents progression phenomenology in CANDU reactors.
- The development of guidelines for the initial response of operators in Control Rooms.
- The development and formation of a support group to assist control room staff during the severe accident evolution.
- The development of flow diagrams for diagnosis.
- The development of operating procedures for severe accidents, including diagnosis, follow-up and long-term monitoring.

During the 2010 / 2013 period the following four sets of guidelines was defined:

- *SACRGs: Severe Accident Control Room Guidelines.*
- *SAGs: Severe Accident Guidelines.*
- *SCGs: Severe Challenge Guidelines.*
- *SAEGs: Severe Accident Exit Guidelines.*

The project is being developed in two stages. The scope for the first stage corresponded to the development of the SAGs, in the following priority order:

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

This stage has been completed and the guidelines are currently being analyzed by the PLEX staff of the CNE.

Besides, the existing SAMGs are being re-evaluated (see section 3.14.3.1.2.3.).

SAMG Training and Validation Exercise conducted by CANDU energy took place at CNE in December 2012.

3.14.3.1.3. CAREM Deterministic Safety Assessment

In order to simulate the plant response in case of anticipated operational occurrences (AOO) and design basis events, a plant model was developed including the primary system (integral type), safety systems and a simplified secondary circuit. The plant code RELAPSIM was used. In order to simulate

CAREM-25 in case of hypothetical severe accidents, reactor and containment models were developed using MELCOR code.

The deterministic safety assessment (DSA) for AOO and design basis events was based on the analysis of more than 40 cases, including those with the failure of the first shutdown system and the success of the second shutdown system. The analysis for most cases was extended to 36 hours in order to cover the grace period of the passive safety systems related to the core cooling function, as well as an analysis that included a station black out (SBO) scenario (considered as design basis event).

In all cases, the respective acceptance criteria are fulfilled with large margins, including those for anticipated transients without scram (ATWS). These acceptance criteria are more restrictive than those internationally accepted for PWRs, as is the LOCA case where it is required that the core remain covered.

DSA for severe accident included the analysis of the reactor and containment in SBO case coincident with a failure of the passive residual heat removal safety system and the largest diameter pipe break (pipe connected to RPV) as well as the passive safety injection system failure. In both cases, a slow progression of the severe accident with heat-up rates lower than 0.1K/s was observed.

3.14.3.2. PROBABILISTIC SAFETY ASSESSMENT

The activities covered during 2010-2012 period in the applications of PSA are the following:

3.14.3.2.1. CNA I Probabilistic Safety Assessment applications

The outcomes of the PSA study for CNA I, have been developed to an enhanced level 1 (L1) according to the objectives of the back-fitting program. This program has added new safety systems, such as the Second Heat Sink (SHS) and redundancy improvements as in the secondary relief valve system.

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 routine tests 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.

ARN adopted an international criterion that limit the increase value of CDF due to design modifications to 1.0×10^{-5} /year maximum. When the allowed value was faced with that of the technical evaluation, it was found that the criterion was not met. As a consequence, the ARN did not authorize such change.

The PSA L1 3rd review corresponding to full power internal events was completed by the Licensee in 2009 and submitted to the Regulatory Body for approval. This PSA 3rd review includes an update of the significant design modifications and incorporates the results of new deterministic studies conducted in the framework of SAMP for CNA I.

These studies extended the modelled 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 will be completed by the second half of 2013. 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 studies used to decide to extend the time between the plant scheduled outages from 12 to 18 months were re-assessed by the Licensee with the mentioned PSA 3rd review, being submitted to the Regulatory Body for approval. However, due to some observations performed by the Regulatory Body, the PSA is yet being reviewed.

The PSA 3rd review 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 plant periodic safety review's safety factors, resulting in the identification of model improvement opportunities as well as the necessary extent of the scope. 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".

NA-SA has begun the update of the L1 PSA corresponding to internal events at full power including the new emergency power supply in the model. It will also be updated the PSA of internal fire. During 2012, walkdowns were conducted to review and update plant fire zones, which results will be included in the mentioned PSA updated version.

The application of PSA results in the safety assessments performed as a response action post Fukushima are the following:

- In order to check the availability of the SSCs which are required in accident scenarios, a SSCs list to check on the walkdowns 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 walkdowns.
- The PSA was used as the main tool to identify the components to be included in the SSEL (see section 3.17.2.3.3.1.1.).

3.14.3.2.2. CNA II Probabilistic Safety Assessment applications

The Regulatory Body has stipulated for the CNA II licensing that the product of frequencies and consequences of severe accidents sequences have to be below certain threshold. The key objective of the PSA level 1 to 3 for CNA II is to demonstrate that the probabilistic requirements of the Argentinean regulation are fulfilled (Regulatory Standard – AR 3.1.3., Radiological Criteria Relating to Accidents in Nuclear Power Plants).

In the 2009-2012 period, the Licensee completed the first PSA L1 revision corresponding to full power internal events. This version was reviewed by the Regulatory Body in 2011 and, as a result, the following main requirements and recommendations were carried out:

- New deterministic studies including LOCAs affecting the moderator system.
- New initiating events not included originally such as high pressure PHTS transients and LOCA's BDBA.
- Modification of some initiating events frequencies according to the accumulated operational experience.
- Re-evaluations to update the frequencies of the loss of external electrical power networks; the service water loss and the shutoff rod system failure rate by using the CNA I operational experience.
- Modifications of the initiating events grouping by excluding from the generic transient some events such as the loss of heat sink which was considered in greater detail in a separate event tree.

The overall result of this study corresponds to a CDF value of $1.9 \cdot 10^{-5}$ / year.

Besides, in parallel to the PSA L1 updated the Licensee has developed the corresponding PSAs L2 and L3. The PSA L2 corresponds to a complete study which includes the responses in the long term of the cooling systems and the containment systems according were foreseen by the design. This study continues the sequences assessment provided by the L1-L2 interface. The grouping was based on the identification of a set of attributes that characterize similar states.

The PSA L2 was also delivered in 2011 and a new revision including the comments from ARN was also accomplished by the utility in April 2013.

The PSA L3 developed uses the results of the PSA L2, which the source term were grouped in material radioactive release categories based on the amount, type and energy of radioactive material and the containment failure path, and assesses the corresponding atmospheric dispersion as well as the resulting dose to the public. Finally, the maximum radiological risk to the public corresponding to each category is calculated to verify the fulfillment of what is established in the Standard AR 3.1.3. - Radiological Criteria Accidents Relating to Nuclear Power Reactors.

The corresponding PSA level 3 was delivered in August 2011 and the revision including the ARN comments was also sent to the ARN in April 2013.

At present the PSA L2 and L3 are being reviewed by the Regulatory Body with the assistance of Sandia N.L. and GRS.

3.14.3.2.3. CNE Probabilistic Safety Assessment applications

In 2004 the Licensee completed a first PSA L1 version considering full power internal events, which 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.). The PSA was applied to demonstrate improvements, in order to ask the Regulatory Body to authorize the corresponding plant modifications.

The impact on the safety due to clogging of heat exchangers caused by the golden mollusc (mejillón dorado) was assessed by using the PSA. Besides, a change in the in-service inspection frequency was decided.

Another application of the PSA was the extended time between outages from 12 to 18 months. As a part of the decision making process, the ARN required to evaluate the impact on the nuclear safety due to a reduction in the frequencies of preventive maintenance, routine tests and in service inspection programs.

As a part of the requirement, NA-SA developed a study using the PSA L1 tool with the plant at full power. Such study fulfils the limiting criterion mentioned in subsection 3.14.3.2.1. The ARN authorized the increased period between outages from 12 to 18 months with a three months tolerance margin.

Currently, as part of the CNE life extension project, the Licensee is developing a PSA L1 revision to update it considering full power internal events, including the design modifications to safety systems planned to be implemented during the refurbishment activities, which is foreseen to be completed by the second half of 2013. In addition to these safety system modifications, the Licensee has new deterministic results corresponding to transients and postulated accidents that allow review some accidental sequences. Preliminarily, the results have let discard some non-conservative hypotheses included in the previous review, such as the use of the moderator as last heat sink in certain LOCA cases that, although until now was considered as effective, according to the new available studies, this cannot be considered as such. Preliminary results for the new revision of the internal events PSA indicate a value of about $2E10^{-5}$ / year for the CDF.

Another important modification introduced in the PSA review refers to the damage states CDS grouping by adapting these states to the definition that requires the continuation of L1 to L2. In this L2 will be used the results and experience in similar CANDU plants, grouping the damage categories according to the characteristics of the consequences to be had in each case. Then, will be identified the contributors sequences to be grouped to each of these damage states coming from different internal initiating events. The CNE PSA L2 is expected to finalize by the end of 2013.

The plant seismic safety assessment will be performed by using a SMA 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, will be identified the SSCs to be included in the SSEL.

Regarding the plant configuration and PSA model, it has been performed a preliminary survey for the future plant configuration (post-PLEX) which includes:

- An important set of design changes to be implemented during the corresponding refurbishment outage. This significantly affects the plant configuration.*
- An update of several hypotheses previously entered in model (Pre-PLEX), which affect in different degrees the new model.*

Therefore, it is expected that from now on, all new PSA studies will be based in the plant configuration and the post-PLEX model which were above mentioned.

3.14.3.2.4. CAREM Probabilistic Safety Assessment applications

In 2010, a L1 PSA within the scope of the basic engineering stage was developed. It was carried out using the technique of large event trees, including, as headings, the active and passive systems involved in the control and mitigation 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. The development of detailed containment event trees is foreseen for 2014.

A PSA L3 was developed where the individual risk to the public was quantified and has been verified that the acceptance criterion was fulfilled with large margin.

3.14.3.3. PERIODIC SAFETY REVIEW

Since 2003 ARN has changed the methodology for license renewal of NPPs in operation, including Periodic Safety Review (PSR) as a licensing tool.

The requirement to develop PSR was included in the operating license of both plants Atucha I and Embalse. It was established that PSR have to be developed each 10 years, with the scope described in the IAEA Safety Standard Series N° NS-G-2.10 "Periodic Safety Review of Nuclear Power Plants" document. Approval of PSR results by ARN is a necessary condition for license renewal.

In the CNA I case, the current license was granted in December 2003, *including the requirement to perform a PSR for its renewal. The scope of the PSR which was agreed with ARN includes the comparison with national and international standards.*

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" have identified new improvement areas. As a result of this analysis an upgrading plan will be sent and agreed with the Regulatory Body.

Other weaknesses, for which the implementation of improvements is not compatible with the remaining plant's life period, are being considered as an input for feasibility analysis for PLEX, which is at present being developed.

A special consideration was given to design changes introduced in the plant since 1990, which represents a much longer period than that taken for most safety factors (in general 10 years). The analysis of all safety related design changes was done, stating the standards taken into account for each one, and the approval process.

Also, the comparison with international standards and practices gave as a result an increase in the resources assigned to activities as ageing management, and equipment qualification, already on going.

The status of probabilistic and deterministic analysis was analysed in specific safety factors. In reference to deterministic analysis, the continuous upgrading of the model of the plant, taking into account new tools and introducing design changes, showed that the design basis had been kept.

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.*

On the other hand, the CNE operating license was emitted in 2007 for a ten years term. However, due to the fact that CNE is engaged in a plant life extension programme, PSR is being developed as part of the safety assessments for that project.

As a condition for renewal of the operating license for the second cycle, ARN requested a PSR with a comprehensive scope which includes an important number of standards for each safety factor. The safety factors of CNE PSR were identified as follows:

1. *Plant design.*
2. *Actual condition of SSCs important to safety.*
3. *Equipment qualification.*
4. *Ageing.*
5. *Deterministic safety analysis.*
6. *Probabilistic safety analysis.*
7. *Hazard analysis.*
8. *Safety performance.*
9. *Use of experience from other plants and research findings.*
10. *Organization, the management system and safety culture.*
11. *Procedures.*
12. *Human factors.*
13. *Emergency planning.*
14. *Radiological impact.*
15. *Severe Accident Management.*

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.

Although the analysis developed for different factors will be sent to ARN following an agreed schedule, the results of the review should be finished 10 months before the plant restart.

3.14.4. REGULATORY PLANT SAFETY PERFORMANCE INDICATORS

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.

3.14.5. COMPLIANCE WITH THE OBLIGATIONS IMPOSED BY THE CONVENTION

The information contained in this section related to probabilistic and deterministic safety assessment of Argentine NPPs in operation, demonstrates that the country complies with the obligation imposed in Article 14 of the Convention of Nuclear Safety.

3.15. ARTICLE 15: RADIOLOGICAL PROTECTION

Each Contracting Party shall take the appropriate steps to ensure that in all operational states the radiation exposure to the workers and the public caused by a nuclear installation shall be kept as low as reasonably achievable and that no individual shall be exposed to radiation doses which exceed prescribed national dose limits.

3.15.1. INTRODUCTION

The control and supervision of NPPs compliance with regulatory standards and other regulatory documents are carried out with a program of routine and non-routine inspections and audits, which contribute to determine the fulfilment of the operating license and any other mandatory documentation.

Resident inspectors carry out this control as well as other technical support (TS) groups belonging to the ARN, who perform independent analyses and evaluations related to different aspects of radiological safety. These TS groups have their own laboratories to perform the measurements and experiments required for this purpose.

Although these control tasks are routinely performed, they are also carried out especially when it is necessary to reinforce the inspection tasks, as in the case of scheduled outages and unforeseen shutdowns or as a consequence of some specific situation, such as an event occurrence.

The periodic test programs related to radiological protection are monitored and observed during their execution, and the experimental data obtained from the mentioned tests are verified by ARN, such as those related to the radiation detection equipment installed in different controlled and supervised areas of the NPPs.

The personnel dosimetry system of the NPPs is evaluated by means of specific audits conducted by ARN specialists. Also, licensee's dosimetry laboratories participate of inter-comparison exercises organized by ARN.

Regarding the control of effluents released to the environment by the NPPs, the surveillance plan during normal operation consists of the measurement of the released activity taken in those places where effluents are emitted. This monitoring plan consists of scheduled routine controls and is complemented with random checks.

In addition to the environmental monitoring plan carried out by the Licensee, ARN performs independent environmental measurements in CNA I and CNE surroundings and nearby zones using its own laboratories and specialists. The control points, where samples are taken, include not only those selected by the installation but also other points chosen according to the ARN criterion.

3.15.2. ATUCHA I NUCLEAR POWER PLANT

3.15.2.1. RADIOACTIVE RELEASES INTO THE ENVIRONMENT

The authorized limits to the environmental releases of CNA I were set *by the Regulatory Body for relevant radionuclides. They were calculated considering the dose, due to all exposure pathways, for a hypothetical critical group.*

The gaseous radioactive releases to the environment due to CNA I operation in the 2010-2012 period may be observed in Table 3.15.1., discriminating those corresponding to I-131, tritium, aerosols and noble gases; it also includes an estimation of the C-14 discharged.

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)
2010	7.1×10^{-5}	5.7×10^2	4.1×10^{-6}	5.0×10^1	5.7×10^{-1}
2011	1.6×10^{-6}	6.1×10^2	1.8×10^{-6}	2.0×10^0	4.8×10^{-1}
2012	5.1×10^{-6}	6.0×10^2	2.1×10^{-6}	1.2×10^1	5.1×10^{-1}

The liquid radioactive releases to the environment by CNA I during 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)
2010	1.1×10^3	1.0×10^{-1}
2011	1.2×10^3	1.5×10^{-1}
2012	1.4×10^3	2.7×10^{-1}

Of the total annual average discharges from CNA I to the environment, *almost 99%* corresponds to tritium.

The CNA I's licensee routinely measures 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" as recommended in the IAEA Safety Reports Series 19 and Safety Series 57. Each model has specific plant information such as critical group's location, habits and food consumption and local dispersion factors of environmental releases.

3.15.2.2. PUBLIC EXPOSURE

The effective doses have been calculated for the most exposed individual considering very conservative hypothesis. The annual dose estimated values are shown in Table 3.15.3., discriminated according to discharge type.

Table 3.15.3. - Critical group individual dose for CNA I

YEAR	GASEOUS DISCHARGE DOSE (mSv)	LIQUID DISCHARGE DOSE (mSv)	TOTAL DOSE (mSv)
2010	5.6×10^{-3}	4.9×10^{-4}	6.1×10^{-3}
2011	5.9×10^{-3}	5.3×10^{-4}	6.4×10^{-3}
2012	4.6×10^{-3}	8.7×10^{-4}	5.4×10^{-3}

3.15.3. EMBALSE NUCLEAR POWER PLANT

3.15.3.1. RADIOACTIVE RELEASES INTO THE ENVIRONMENT

The authorized limits to the environmental releases of CNE were set *by the Regulatory Body for relevant radionuclides. They were calculated considering the dose, due to all exposure pathways, for a hypothetical critical group.*

The gaseous radioactive releases by CNE to the environment, for the *2010-2012* period may be seen in Table 3.15.4., discriminating those corresponding to I-131, Tritium, aerosols and noble gases; and including an estimation of C-14 discharges.

Table 3.15.4. - Activity released from CNE to the environment as gaseous discharges

YEAR	I -131 (TBq)	TRITIUM (TBq)	AEROSOLS (TBq)	NOBLE GASES (TBq)	C -14 (TBq)
<i>2010</i>	<i>7.4×10^{-7}</i>	<i>2.9×10^2</i>	<i>8.7×10^{-6}</i>	<i>4.1×10^1</i>	<i>3.7×10^{-1}</i>
<i>2011</i>	<i>3.8×10^{-6}</i>	<i>3.1×10^2</i>	<i>1.5×10^{-3}</i>	<i>9.2×10^1</i>	<i>3.4×10^{-1}</i>
<i>2012</i>	<i>4.1×10^{-6}</i>	<i>3.6×10^2</i>	<i>1.5×10^{-4}</i>	<i>2.8×10^1</i>	<i>3.3×10^{-1}</i>

The liquid radioactive discharges to the environment by CNE, 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 CNE to the environment as liquid discharges

YEAR	TRITIUM (TBq)	OTHER RADIONUCLIDES (TBq)
<i>2010</i>	<i>1.2×10^2</i>	<i>8.6×10^{-3}</i>
<i>2011</i>	<i>1.3×10^2</i>	<i>5.2×10^{-3}</i>
<i>2012</i>	<i>4.8×10^2</i>	<i>2.1×10^{-2}</i>

As in the case of CNA I, *91%* of the total annual average discharges from CNE to the environment, correspond to Tritium.

The CNE's licensee routinely measures 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" as recommended in the IAEA Safety Reports Series 19 and Safety Series 57. Each model has specific plant information such as critical group's location, habits and food consumption and local dispersion factors of environmental releases.

3.15.3.2. PUBLIC EXPOSURE

The annual critical group doses due to CNE's operation during the *2010-2012* period are presented in Table 3.15.6., discriminated according to discharge type.

Table 3.15.6. - Critical group individual dose for CNE

YEAR	GASEOUS DISCHARGE DOSE (mSv)	LIQUID DISCHARGE DOSE (mSv)	TOTAL DOSE (mSv)
<i>2010</i>	<i>9.2×10^{-4}</i>	<i>4.1×10^{-3}</i>	<i>5.0×10^{-3}</i>
<i>2011</i>	<i>1.0×10^{-3}</i>	<i>3.5×10^{-3}</i>	<i>4.5×10^{-3}</i>
<i>2012</i>	<i>8.4×10^{-4}</i>	<i>1.2×10^{-2}</i>	<i>1.3×10^{-2}</i>

The main contributor to the critical group's dose corresponds to the liquid discharges due to the fact that they are released into the Embalse Lake.

The increase in the dose during 2012 was mainly due to a loss of heavy water from a steam generator of the plant.

3.15.4. 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, as well as 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.

3.15.4.1. DOSE LIMITS TO WORKERS

The Regulatory Standard AR 10.1.1. establishes that the dose limits have not been exceeded when the following conditions are fulfilled:

$$\frac{H_p(d)}{L_{DT}} \leq 1$$

and

$$\frac{H_p(10)}{20 \text{ mSv}} + \sum_j \frac{I_j}{I_{L,j}} \leq 1$$

Where:

$H_p(d)$ is the personnel equivalent dose at a depth of 0.07 mm and 3 mm (for skin and crystalline respectively), integrated in a year,

L_{DT} is the 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 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.

3.15.4.1.1. Occupational dose at CNA I

The collective effective dose and the average individual effective dose received by workers in CNA I during the 2010-2012 period, are presented in Table 3.15.7.

Table 3.15.7. - Occupational dose in CNA I

YEAR	COLLECTIVE EFFECTIVE DOSE (man Sv)	AVERAGE EFFECTIVE DOSE (mSv)
2010	1.1	1
2011	4.6	3
2012	3.2	2

The differences in annual collective effective doses are related to the programmed outages. *For example, in 2010 there wasn't any programmed outage at CNA I.*

3.15.4.1.2. Occupational dose at CNE

The collective effective dose and the average individual effective dose received by CNE workers during the 2010-2012 period are presented in Table 3.15.8.

Table 3.15.8. - Occupational Dose in CNE

YEAR	COLLECTIVE EFFECTIVE DOSE (man Sv)	AVERAGE EFFECTIVE DOSE (mSv)
2010	3.5	3
2011	2.1	2
2012	1.9	2

Occupational doses in CNE are lower than those recorded in CNA I due to the technological differences between both NPPs as well as to the longer operation period of CNA I compared to CNE.

3.15.5. ALARA ACTIVITIES

ALARA program is carried out in both 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.
- operating experience feedback.
- mock-up training for the above mentioned activities.
- design of specific tools and shielding to be used in high radiation fields.

The ALARA activities have been reinforced and intensified with the objective of optimizing the dose received by personnel occupationally exposed to ionizing radiations, reinforcing its activity with better implementation of mock-ups, training, improvement in tools and the analysis of work plans.

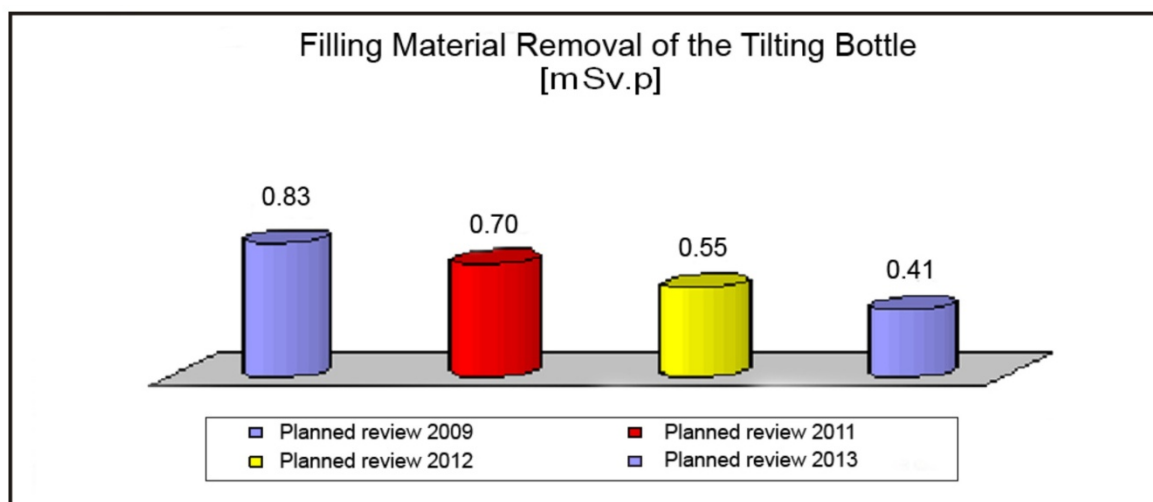
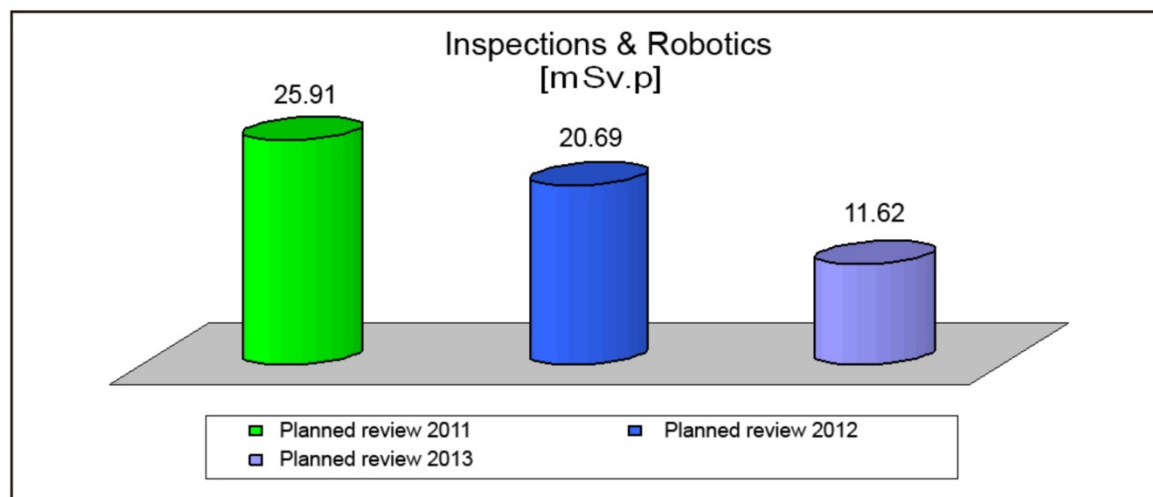
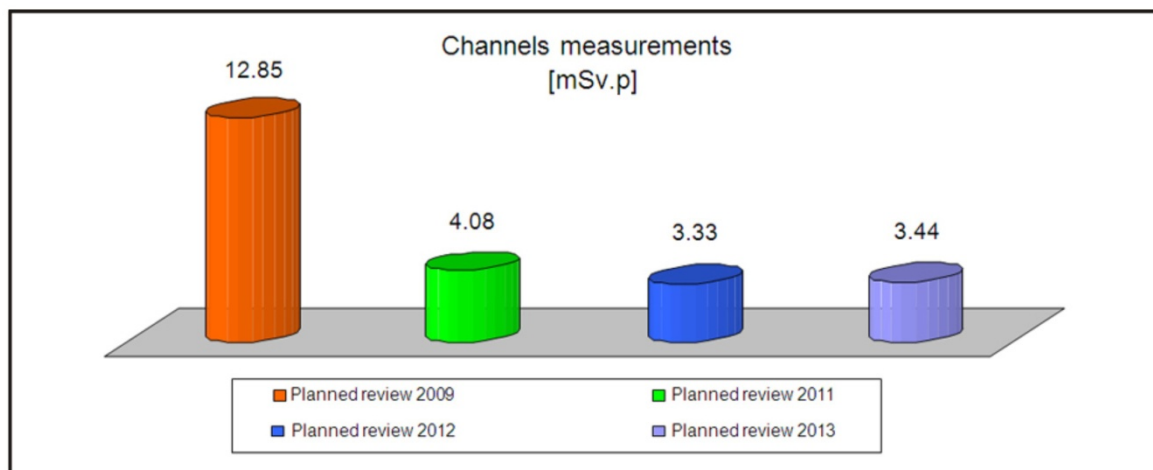
The charts showed in 3.15.5.1., 3.15.5.2. and 3.15.5.3. present some examples of dose improvements for repetitive tasks corresponding to CNA I y CNE.

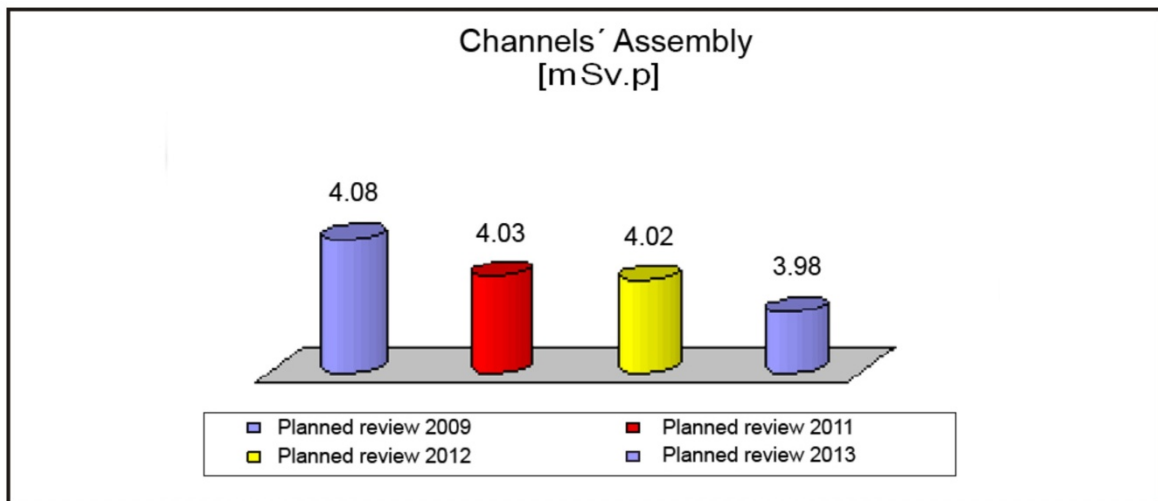
3.15.5.1. CNA I REACTOR INSPECTION PROGRAM

The reactor inspection program requires performing, at each scheduled outage, the metrology of the primary coolant channels (that are transported to the spent fuel storage pool building I) and inspections of the reactor internal components. Dose reduction achieved for the different scheduled outages which is shown below, is due to:

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

Below are showed comparison charts for different tasks and sub-tasks to carry out the inspection program.



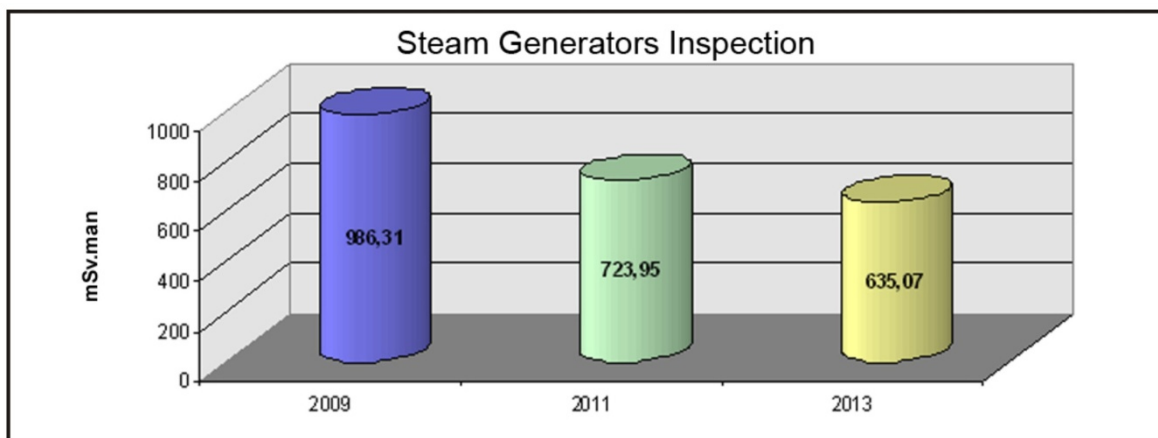


3.15.5.2. CNA I STEAM GENERATORS INSPECTION

Dose reduction achieved for different scheduled outages due to:

- *Shorter exposure times in areas with high radiological exposition.*
- *Better control and monitoring of the personnel individual dose. Use of tele-dosimetry.*
- *Involved personnel highly qualified.*
- *Location changing of the racks to less exposed areas (in previous plant scheduled outages they were in front of the boiler plenum. At this scheduled outage they were placed at the main pumps building).*

The figure below shows the corresponding comparison chart.



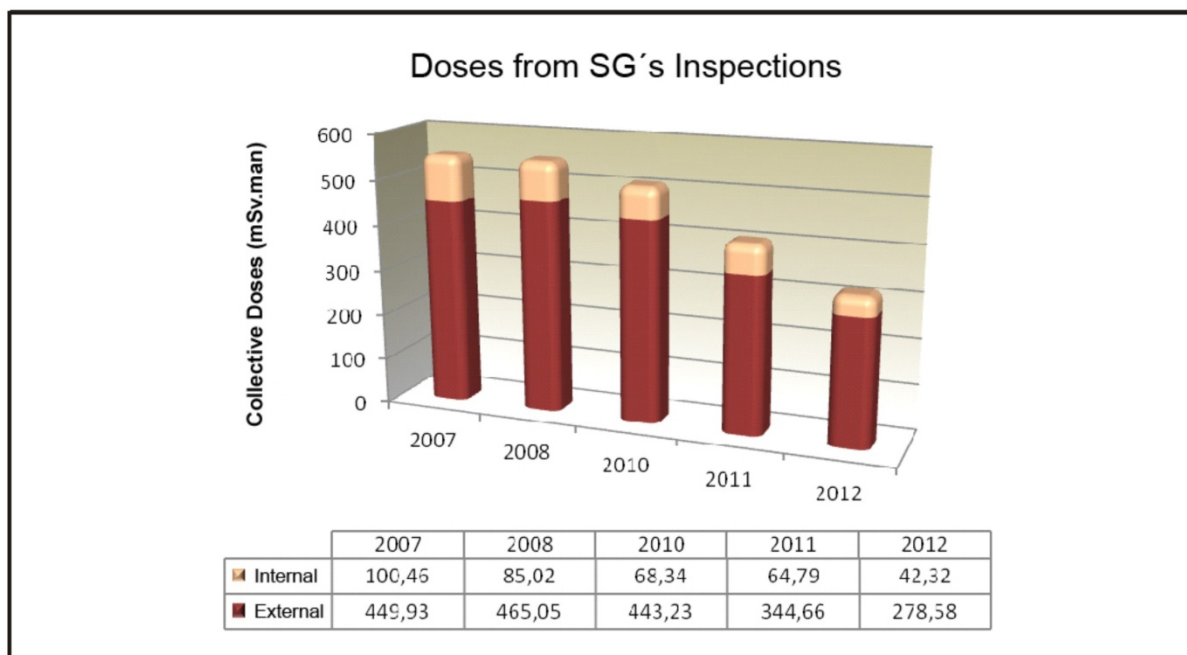
3.15.5.3. CNE STEAM GENERATORS INSPECTION

The optimization of this task has been achieved by mean of a sustained doses reduction through the last five plant scheduled outages, in which the performed works had been the same. Important factors that contributed to the dose reduction were the following:

- *New vacuum for boiler plenum cleaning (by reducing γ doses +pollution).*
- *Reduced maintenance times on tele-manipulators (reduction of γ doses).*
- *Implementation of bypass system for probes calibration, avoiding manual calibration on the SGs cabinet (reduction of γ doses+H3+pollution).*
- *Increases the decontamination in SG cabinets and mini-zones (reduction of γ doses).*
- *Increased use of plastic suits (reduction of H3 + Pollution).*

- Better control and monitoring of the staff's individual doses of sealing, welding and non-destructive test (reduction of γ doses).
- Improved in the SGs' vacuum during its opening (reduction of H3 + Pollution).
- Improved communication and video systems (reduction of γ doses+H3+pollution).
- Optimal training of inspection and radiological assistance staff.
- Use of tele-dosimetry system since 2011 scheduled outage.

The figure below shows the corresponding comparison chart.



3.15.6. COMPLIANCE WITH THE OBLIGATIONS IMPOSED BY THE CONVENTION

The ALARA activities have been reinforced and intensified with the objective of optimizing the dose received by the personnel occupationally exposed to ionizing radiations, reinforcing its activity with the better implementation of mock-ups, training, improvement in tools and the analysis of the work plans. Limits for the doses for personnel occupationally exposed have not been exceeded. The annual average value of doses for the public (critical group) resulted lower than 2.4% of the established individual dose constraint established for NPPs design which is 0.3 mSv.

Therefore the country complies with the obligations imposed in Article 15 of the Convention on Nuclear Safety.

3.16. ARTICLE 16: EMERGENCY PREPAREDNESS

- i. *Each Contracting Party shall take the appropriate steps to ensure that there are onsite and off-site emergency plans that are routinely tested for nuclear installations and cover the activities to be carried out in the event of an emergency.*

For any new nuclear installation, such plans shall be prepared and tested before it commences operation, above a low power level agreed by the Nuclear Regulatory Authority.

- ii. *Each Contracting Party shall take the appropriate steps to ensure that, insofar as they are likely to be affected by a radiological emergency, its own population and the competent authorities of the States in the vicinity of the nuclear installation are provided with appropriate information for emergency planning and response.*

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

The Regulatory Body requires from the Licensee a plan to respond in case of a nuclear or radiological emergency, to be applied inside and outside the NPPs. This plan, usually known as "Internal and External Emergency Plan", shall comprise every aspect related to the strategy required in order to control and limit the accident's consequences. This plan is part of the documentation required by ARN to grant the NPP's operating license.

In November 1998, the Executive Branch subscribed the Decree N° 1.390 defining the scope and procedures that facilitates the enforcement of Act N° 24.804 (National Law of Nuclear Activity). The application of this Decree establishes the functions of ARN related to nuclear emergencies, preparedness, training and response.

The main important issues are:

- ARN is empowered to regulate and control the nuclear activity as well as to advise the Executive Branch on issues about its scope, including radiological and nuclear emergencies.
- ARN must provide protection against harmful effects of ionizing radiation, even under emergency situations.
- ARN must set the guidelines and criteria for the emergency plans and training of the public members, near NPPs, in the case of nuclear accidents.
- Emergency plans must be developed by local, provincial and national authorities; they must contemplate an active participation of the entire community and be approved by ARN.
- ARN must conduct the actions within the area covered by the emergency plans. Law Enforcement entities and representatives of Civil Organizations must *respond* to the designated ARN officer.

Furthermore, emergency exercises are carried out annually (explicit demand in the NPP's operating licence), aiming at the evaluation of the response team and the subsequent improvement of the plan. The specific objectives of each annual exercise must be agreed on with ARN.

The main objectives of the emergency exercises are:

- To extend the diffusion of the current procedures among the response organizations.
- To establish command systems and verify the capacity of application of the radiological precautionary actions (residential sheltering, evacuation, thyroid prophylaxis, access controls, notifications, messages to the population, etc.).
- To involve the members of the public with planned actions and their *simulation*.

- *To promote that each response organization intervening in an emergency has an updated procedure according to its function.*
- *To keep updated the training of the response organizations and the local community.*
- *To identify needs and weaknesses to incorporate them as opportunities of improvements.*

3.16.2. EMERGENCY PLANS AND PROGRAMMES

3.16.2.1. ARN RESPONSIBILITIES RELATED TO NUCLEAR EMERGENCIES

ARN continues dealing with prevention of accidents at nuclear installations as its regulatory function.

The criteria that must be adopted by the Licensee are established in the ARN document “Criteria for the intervention in nuclear emergencies with off-site radiological consequences”. Its compliance has been required of CNA I, CNE, and must be applied in CNA II, which as were previously mentioned is in the *commissioning* stage.

In order to accomplish what is set in Act N° 24.804 and Decree N° 1.390, ARN's - Nuclear Emergency Response System (SIEN) was created by ARN Resolution N° 25 on November 1999.

The SIEN coordinates the national, provincial and local response organizations (National Civil Protection Direction, Provincial Civil Defence and Local Civil Defence of every Municipality within a 10 km radius around each NPP) to effectively manage nuclear emergencies in preparedness, intervention and recovery stages.

In particular, the operational capacity of the ARN in the management of the implementation of actions to protect the members of the public in the surroundings of the NPPs from the radiological consequences is *periodically* checked by carrying out the emergency exercises.

ARN, in addition to its main role as head of the Incident Command System for off-site consequences, performs the nuclear and radiological assessments, the radiological protection of intervening teams and the environmental monitoring. Representatives of all the intervening organizations (as established in the Emergency Plan) integrate a Response Command System and ARN coordinates the response teams that belong to civil organizations (Fire fighter brigades, Civil Defence, etc.), law enforcement (Police, Gendarmerie and Coast Guard), and military institutions (Army, Navy and Air Force). These organizations apply the precautionary measures with their response teams. All these groups have procedures to deal with nuclear emergencies, under ARN coordination.

A Nuclear Emergency Operative Chief (JOEN) from the ARN is designated and integrated to the *Municipal* Emergency Operative Centre (COEM), in order to conduct the actions within the 10 km radius established as the “precautionary action zone” (*PAZ*). The ARN-JOEN is the officer to whom civil organizations and law enforcement must report to.

An Emergency Control Centre (CCE) has been set up at ARN's Headquarters in order to coordinate the response and support to the ARN representative in the COEM. ARN's strategy to respond to a nuclear emergency consists in establishing expert teams and a decision making team at Headquarters.

ARN is the National Competent Authority according to the “Convention on Early Notification of a Nuclear Accident” and in the “Convention on Assistance in the case of a Nuclear Accident or Radiological Emergencies”, and the CCE is the National Warning Point according to the IAEA - Emergency Notification Assistance Technical Operations Manual (ENATOM).

During the accident occurred on 11th March 2011 at Fukushima, Japan, the ARN's SIEN made a daily tracing of the actions taken by the manager responsible of the emergency response. This circumstance allows making tests and desktop exercises to compare the actions taken in Fukushima with the eventual actions that would be carried out in Argentina on a similar situation. These actions were taking into account in the NPPs' emergency plans. At the same way it was made a tracing of the corresponding reports from medical point of view and the protection measures taken were considered.

As a consequence of the Fukushima accident and with the purpose of apply the corresponding lessons learned, the Regulatory Body requested perform a stress test to each Argentinean NPP (see Annex VIII) 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) *Aspects related to severe accidents management.*
- d) *Internal management of the emergency, including emergency management and control; fuel damage mitigation; revision of procedures; personnel training and equipment.*
- 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 Licensee of CNA I; CNA II and CNE 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 this reports to verify compliance with the provisions of the regulatory requirement. As a result many opportunities for improvements have been identified but, in no case were showed weaknesses or critical situations (cliff edge) which make necessary to take urgent actions.

The improvements and modifications proposed by the NPPs Licensees included an implementation schedule composed by short, medium and long term actions; the latter reaching the year 2016. Finally, as 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 containing numerous improvements and modifications to be implemented by the Argentinean NPPs, some of which have already been implemented.

3.16.2.2. REQUIREMENTS CONCERNING EMERGENCY PLANS

ARN has regulated the preparedness and response to emergency situations in NPPs, through different mandatory documents, such as the Regulatory Standards AR 10.1.1. and AR 3.7.1., Operating Licenses and specific requirements to the Licensee and Primary Responsible of the installations.

The conditions to be fulfilled by the NPP's emergency plans are the following:

- The implementation of protective actions of automatic application must be foreseen within a circular area of a radius of 3 km *with centre on the installation* that is called Emergency Zone.
- Once the accident has occurred, a set of protective actions established in the emergency plan must be taken.

The protective actions carried out on the basis of the plant situation and meteorological conditions (before carrying out environmental monitoring) in the Precautionary Action Zone (PAZ), defined as the area enclosed in a 3 km radius in all directions around the NPP as well as an area up to 10 km within an angle of 60 degrees in the direction of the wind, are:

- Early evacuation within the 3 km radius in all directions.
- Access Control as defined in the Emergency Plan.
- Sheltering: people must be informed through the massive communication media.
- Iodine Prophylaxis: distribution of stable iodine pills to the involved people must be implemented. The iodine pills must be swallowed one per day, by every person staying within the above mentioned key-hole, including milk-fed babies, children and pregnant women (*dose according to their age*).

The environmental monitoring starts once the radioactive material release has ended. The environmental monitoring results are used to define an Urgent Protective Zone (UPZ), where the following protective actions are taken (see Table 3.16.1.):

- Evacuation from the zones affected by radioactive deposit. This action must be implemented:
 - Every time the radiation level of the material deposited on the ground is equal or higher than 100 mSv, integrated during the first 6 hours from the radioactive emission.
 - Optionally, if the same dose (100 mSv) is integrated during the first 24 hours from the accident.
- Resettlement evacuated zones: values were calculated in the 80's and are currently being reviewed.

- Intervention in relation to food: the intervention levels adopted by the ARN for the substitution of contaminated foodstuffs for consumption were obtained from an optimizing analysis in which the expected effects of such food consumption and the drawbacks produced by their absence or replacement by non-contaminated products were taken into account. Due to the country's characteristics, the contaminated food may generally be replaced by other products coming from areas not affected by the accident, which means that the mentioned intervention levels are significantly lower than those from other countries (see Table 3.16.2.).
- Decontamination of ground: due to its high cost the implementation of this action must be decided on a case-by-case analysis.
- Iodine prophylaxes.*

Table 3.16.1. - Protective measures and intervention levels

	PROTECTIVE ACTION	APPLICATION	INTERVENTION LEVELS	COMMENTS
URGENT PROTECTIVE ACTIONS	Early evacuation within a 3 km radius	Always	n.a.(*)	Protective measures carried out on the basis of the plant situation and meteorological conditions.
	Access control and road blocks	Always	n.a. (*)	
	Sheltering	Always	n.a. (*)	
	Iodine prophylaxis	Always	n.a. (*)	
NON URGENT PROTECTIVE ACTIONS	Evacuation	Always	100 mSv (**)	Dose integrated during the first 6 h after the radioactive deposition.
		Optional	100 mSv (**)	Dose integrated during the first 24 h after the radioactive deposition.
	Resettlement of evacuated zones	Always	100 mSv (**)	Dose integrated in one year.
	Land decontamination	Optional	n.a. (*)	This action shall be decided on a case-by-case analysis.
	Food ban	Always		Table 3.16.2.

(*) Non-applicable.

(**) Values calculated in the 80's, today in review.

Table 3.16.2. - Intervention levels for foodstuff, in Bq/kg, as a result of a generic optimization analysis

Group	Cereals	Vegetables Green	Vegetables	Fruits	Meats	Milk	Fish
1	10	10	15	15	50	10	10
2	1000	1000	1500	1500	5000	1000	5000

Group 1: ²³⁸Pu, ²³⁹Pu, ²⁴⁰Pu, ²⁴¹Am, ²⁴²Cm, ²⁴⁴Cm, ²³⁹Np

Group 2: ¹³⁷Cs, ¹³⁴Cs, ¹³¹I, ⁸⁹Sr, ⁹⁰Sr, ⁹⁵Zr, ¹⁰³Ru, ¹⁰⁶Ru, ¹⁴⁰Ba, ¹⁴⁴Ce

3.16.2.3. CLASSIFICATION OF EMERGENCY SITUATIONS

The emergency is classified as:

- General Emergency: Events resulting in an actual or significant risk of a release requiring implementation of urgent protective off-site actions.
- Site Area Emergency: Events resulting in a major decrease in the NPP defence in depth level.
- Alert: Events involving an unknown or significant decrease in the NPP defence in depth level.

When an abnormal situation arises at a NPP, the Primary Responsible (NPP'S manager) must classify the emergency, apply the plant emergency procedures and notify the external emergency *response* organizations and the ARN.

These emergency procedures define two emergency states:

- Green Alert: when an abnormal situation that compromises the nuclear safety is detected. The internal emergency procedures are applied and the internal emergency organization is established. Without any delay, the Primary Responsible must start implementing *automatic* protective actions (messages to population provide Iodine prophylaxis, preparation for access control, sheltering, and evacuation in the 3 km radius around the NPP). At the same time, the Primary Responsible requests the formation of the external emergency *response* organization (COEM).
- Red Alarm: it is declared by the Primary Responsible when the emission of significant amounts of radioactive material is initiated. Internal emergency procedures continue and the situation implies notifying to the external emergency organization (COEM) who immediately communicates the Red Alarm and applies protective actions (iodine prophylaxis, sheltering, access control, evacuation, consumption restrictions, etc.).

During the On-site and Off-Site emergency, COEM and the Primary Responsible maintain an exchange of information.

ARN warns the National Response System when the Red Alarm is declared. For the international notification, the Convention on Early Notification of a Nuclear Accident format is adopted, and ARN is the National Competent Authority who declares and notifies the alarm to IAEA and the potentially affected neighbouring countries.

3.16.2.4. ON-SITE AND OFF-SITE NPP EMERGENCY PLANS

The NPP's internal and external emergency plans include the necessary information for preparedness and response in case of nuclear emergencies.

Furthermore, these emergency plans take into account the following intervention objectives:

- To lead the accidental situation.
- To assess the potential consequences.
- To notify the corresponding emergency situation.
- To implement the protective action in order to avoid or mitigate radiological consequences on the individuals and the environment.
- To apply the necessary actions to restore the liveability conditions in the involved zone.

The NPPs' emergency plans of both NNPs in operation (CNE and CNA I) include:

- Concerning Organization and Responsibilities:
 - Agreements with intervening organizations in order to implement the protective actions.
 - Responsibilities and functions of the organizations during the response.
 - The arrangement, responsibilities and specific function of the NPPs Internal Committee of Emergency Control.
- Concerning the procedures, they include:
 - Different alert levels:
 - Internal onsite alert.
 - External offsite alert.
 - Onsite emergency.
 - Offsite emergency.
 - The correspondence between the different emergency levels and the alarm levels of Civil Defence.

- The following actions to face an emergency situation:
 - Quick emergency detection.
 - Emergency Response System activation.
 - Evaluation of the situation.
 - Start and stop of the protective actions.
 - Remedial actions.
- The communication systems protocols necessary to the emergency management.
- The detailed protective actions, bearing in mind the accident and its possible evolution, mainly for the following cases:
 - Noble gases release only.
 - Noble gases and volatile elements release.
 - Noble gases, volatile elements and aerosol release.
- The way of implementing protective actions, including:
 - The circumstances in which they must be implemented.
 - Organizations in charge of their implementation.
 - The zone where they must be implemented.
- Protocols for alert communication, information and instructions to the potentially affected population (broadcasting, television, loudspeakers, alarms, etc.).
- Protocols of dose control for the emergency response team.
- Concerning places and equipment:
 - Sets the sites for the evacuated people up to a radius of 3 km around the NPPs.
 - Specify the assembly point in the case of a possible evacuation, which shall be also able to give eventual sheltering.
 - Establish the necessary equipment that must be available for radiological monitoring,
 - Establish On-site and Off-site location for the NPP Internal Committee of Emergency Control.
 - Define who and where the public and media must be informed.
- Concerning maintenance of resources, they contain:
 - A procedure to update and upgrade the emergency plan.
 - A calibration and maintenance program for emergency equipment.
 - An annual exercise of the emergency plan application.

3.16.2.5. NATIONAL NUCLEAR EMERGENCY PLAN

ARN is a National organization that acts in cases of nuclear *and radiological* emergencies, as described above, and works together with different national organizations (Law Enforcement, National Civil Defence, National Meteorological Service, National Hospitals and Military Institutions). Many of these national organizations have been cooperating in emergency preparedness, response and training activities before the Act Nº 24.804 was sanctioned.

As was stated previously in the National Nuclear Safety Reports, there wasn't a contingency plan in the national scope until 1998, when a Project for a "National Civil Protection Program" was established which included nuclear emergencies and sets the basis for the Federal Emergency System which is described below.

At the end of 1998, the National Government started the organization of the Federal Emergency System to ensure the co-ordination among all national organizations and to interact with local and province response organizations. ARN and CNEA are members of this Federal Emergency System.

In October 1999, the Executive Branch sanctioned Decree Nº 1.250 establishing the Federal Emergency System, which does not replace the function of any other response organization at different levels, but ensures that there are no overlapping responsibilities during an accident. In 2005,

by Decree N° 1.067, the National Civil Protection Direction (DNPC) assumes the functions of the Federal Emergency System and became the highest coordinating structure at national level.

The National Civil Protection Direction, as a national structure, covers the organizational needs concerning nuclear emergencies. The National Contingency Plan for Nuclear Emergencies developed together by the ARN and the DNPC staff was concluded and approved by ARN.

3.16.2.6. NUCLEAR EMERGENCY PLAN AT PROVINCE SCOPE

Civil Defence of Buenos Aires and Córdoba are involved in nuclear emergency exercises together with the local Civil Defence. Its function is to assist the local civil defence with resources and trained staff. The ARN has approved the Nuclear Emergency Plan of Córdoba and the Nuclear Emergency Plan of Buenos Aires is being developed.

3.16.2.7. NUCLEAR EMERGENCY PLAN AT LOCAL SCOPE

The *NPPs'* emergency plans are compatible with every Local Emergency Plan within 10 km around each NPP. ARN approves these plans and they are tested by means of exercises at least once a year, in accordance to licensing conditions, in which all the organizations and public involved participate. The exercises performed till now were satisfactory, taking into consideration the proposed objectives.

3.16.2.8. IMPROVEMENTS CONCERNING EMERGENCY PREPAREDNESS IN THE ARN

Geographical Information System (GIS)

Since 2003, the Emergency Control Centre of the ARN has a Geographic Information System (GIS) as a versatile tool for planning the nuclear emergency response.

The GIS is composed of a database with social, economic, physical and environmental data and incorporates results for forecasting models and measurements of environmental impact during the emergency. This enables the analysis of data in real time and facilitates decision-making actions.

The application of GIS in emergencies contributes to establish feasible scenarios, the definition of the most likely ones, and the determination of their possible effects. Thus, for each scenario the most appropriate actions may be analysed and pre-establish them for each particular area, *for example:* alternative evacuation routes, *reception place for evacuees*, action of the response units and location of the monitoring teams *operation areas for aerial monitoring, persons and vehicles decontamination areas, etc.*

The results of the consequences assessment models, which are added into the GIS for further analysis and integration with the current database, correspond to the results of Dose Assessment System Accident (SEDA), *developed by the ARN*, and International Exchange Program (IXP of U.S. National Atmospheric Release Advisory Centre, NARAC, for the National Nuclear Security Administration, NNSA, U.S. Department of Energy).

The first model is a calculation code based on a Gaussian atmospheric dispersion model applicable to releases of radioactive materials in a NPP. Its input data are: meteorological data (speed, wind direction and intensity, cloud cover or vertical temperature gradient and atmospheric stability class) and data from the release of contaminants (qualitative and quantitative estimate of the radionuclide's released into the atmosphere). The results are isoconcentration and isodose lines that define the scope of protective actions.

The second model is a prediction system for real-time atmospheric transport of radioactivity from a nuclear accident. It is based on Web-Full Version of NARAC computer system, that users employ around the world to perform time-dependent atmospheric dispersion and dose calculations. It has real-time data from a large number of weather stations that allow knowing the current weather conditions in the point of interest, as well as the forecast for the next few hours. With this information and data release of pollutants, different models can be used to estimate their impact. The results obtained are: exposure rate contours, isolines of activity of radioactive material deposited on the ground; recommendation of protective actions.

These new tools are incorporated into the CCE and contribute to improve the Emergency Preparedness and Response System.

Emergency Control Centre

The ARN Emergency Control Centre (ECC), located at ARN Headquarters, has upgraded the communications system. An independent Local Area Network has been installed. A GIS server has been installed able to transmit processed information at the ECC, through Internet, to COEM. It *counts on* an independent Local Area Network, able to transmit videoconferencing *through* internet and ISDN. In addition, a satellite system allows the sending of information (video, data and voice) from the environmental monitoring location to the ECC.

To improve the communication systems ARN worked altogether with the NPP's emergency responsible, unifying communication via satellite. In this way, currently both ARN and the NPP's emergency responsible work with the same technology.

For the next stage it is foreseen to establish a point-to-point connection with the NPPs. It would allow the transmission of on-line information about the reactor data and environmental measurements in emergency situations, and videoconferencing in a more efficient way.

3.16.2.9. IMPROVEMENTS CONCERNING EMERGENCY PREPAREDNESS IN THE NPPS

The analysis and assessments, as part of the stress tests carried out in 2011 / 2012, required by the Regulatory Body related with emergencies management and control, fuel damage mitigation, radioactive emissions reduction, revision of procedures, personnel training and equipment availability were performed. The improvements proposed by the Licensee as the stress tests results include the following:

3.16.2.9.1. CNA I

- *An environmental monitor net around the plant which consists of mobile and fixes radiological and meteorological stations was implemented to facilitate an efficient response in accidental situations by having real time information.*
- *Source term:*
 - *A verification of the reactor and the spent fuel pools source term is being performed.*
 - *Update the source term of the calculated scenarios with radiological consequences for all conceivable types of severe accidents at the site. This improvement will be carried out when the PSA level 2 be available.*
- *Suitable amount of new fix and mobile satellite phones, available to be used as a back-up of the existing communication systems, were provided.*
- *Procurement of a foam and air system against fire (cannon for long distance action) installed in a mobile foundation in the heliport zone which can eventually contribute to the washing and the retention of radioactive particles is under way.*
- *Revision of current procedures applicable to the strategy of emergency management, including fire fight and the corresponding recuperation actions considering the lessons learned from Fukushima. The implementation of this improvement is foreseen by 2014.*
- *Modify the PS-101 procedure "Conformation and function of the control of emergency internal centre – CICE" to include the CNA II plant staff. The implementation of this improvement is foreseen by the end of 2013.*
- *Conditioning of a municipal emergency control centre far from the planning zone of the nuclear emergency (more than 10 km). The centre must have infrastructure, instrumentation and communication systems according to what was learnt in Fukushima accident:*
 - *The Licensee was required to submit for approval to the Regulatory Body a proposal detailing the features of the corresponding infrastructure, instrumentation, and the communication systems. The implementation of this improvement is foreseen by the end of 2014.*
 - *The approved municipal emergency control centre must be implemented according to the features indicated in the previous bullet. The implementation of this improvement is foreseen by the end of 2015.*

- *Emergency exercises extended in time are being planned. The exercises aim to have operative emergency centres during a long term to improve the emergency direction transfer mechanism, the resources sustainability, the technical support provision, the communications and trustworthy data acquisition. The implementation of this improvement is foreseen by the end of 2014.*
- *Relocation of the Emergency Control Internal Centre (CICE) in the new unique lodge for the site will have secure energy supply, a system of positive pressure ventilation and a filter chain to allow extended stays of the emergency conduction personnel inside this building. The implementation of this improvement is foreseen by the end of 2015.*
- *Analysis of different facilities to retain the water used in decontamination tasks is under way.*
- *A new building will be constructed to house a meeting point with a capacity of 1,300 persons; it will have secure energy supply, a system of positive pressure ventilation with a filter chain allowing that the personnel can stay inside this building for a long term. The implementation of this improvement is foreseen by the end of 2016.*

3.16.2.9.2. CNA II

- *An environmental monitor net around the plant was implemented (idem for CNA I).*
- *Suitable amount of new fix and mobile satellite phones foreseen by the end 2013.*
- *To dispose of an Emergency Plan which includes the lessons learned from Fukushima related with the management and control, fuel damage mitigation, reduction of radioactive emissions, revision of procedures and equipment. The implementation of this improvement is foreseen before the nuclear commissioning beginning.*
- *Dispose of a municipal emergency control centre far (more than 10 km) from the planning zone of the nuclear emergency (idem for CNA I).*
- *Relocation of the Emergency Control Internal Centre (CICE) (idem for CNA I).*

3.16.2.9.3. CNE

- *An environmental monitor net around the plant which consists of mobile and fixes radiological and meteorological stations was implemented to facilitate an efficient response in accidental situations by having real time information.*
- *Source term:*
 - *A verification of the reactor and the spent fuel pools source term is under way.*
 - *Update the source term of the calculated scenarios with radiological consequences for all conceivable types of severe accidents at the site. This improvement will be carried out when the PSA level 2 be available.*
- *Suitable amount of new fix and mobile satellite phones (idem for CNA I).*
- *The re-evaluation of all the current procedures applicable in the emergency management strategy is taking place including the evaluation of the need to generate other procedures. The implementation of this improvement is foreseen by the end of 2014.*
- *For attend the needs to perform different works outside the Service Building, it was foreseen to incorporate:*
 - *Three 5.5 kW portable electric generators with light columns.*
 - *Three 6 kW mobile diesel generators that are located in trailers with its mobile light columns.*

The implementation of these improvements is foreseen by the end of 2013.

- *An electric panel will be installed outside the service building with connection facilities to the emergency light equipment; powered by batteries (Class I). The implementation of this improvement is foreseen by the end of 2015.*
- *The review and improvement of the procedures applied to the co-ordination and reception of supplies, equipment and additional personnel involved in the emergency management will be implemented by the end of 2014.*

- *Emergency exercises extended in time are being planned. The exercises aim to have operative emergency centres during a long term to improve the emergency direction transfer mechanism, the resources sustainability, the technical support provision, the communications and trustworthy data acquisition. The implementation of this improvement is foreseen by the end of 2014.*
- *The construction of small trenches with plastic membranes that make them waterproof to retain the water used in decontamination tasks has been foreseen by the end of 2015.*
- *Conditioning of a municipal emergency control centre far from the planning zone of the nuclear emergency (more than 10 km) (idem for CNA I). The implementation of this improvement is foreseen for the end of 2016.*
- *The Emergency Internal Control Centre (CICE) building will be modified and improved regarding seismic aspects. A ventilation system with HEPA / active carbon filters and emergency electric energy will be installed. In addition to these, the communication system will be improved to stand severe accident conditions for long term. The implementation of these improvements is foreseen for the end of 2016.*
- *Installation of an air recirculation filter system in the secondary control room (SCR) that allows its habitability when a smoke or radioactive emission to the atmosphere occurs during the nuclear accident. The SCS ventilation system will be improved installing a 100% air recirculation system by means of active carbon filters and absolute filters. The systems will be powered from the secure electric supply system (EWS). The implementation of this improvement is foreseen by the end of 2016.*

3.16.2.10. ACTIONS TAKEN FOR THE IMPROVEMENT OF THE EMERGENCY EXERCISES

During the considered period, it was considered necessary to qualify the emergency response groups from all radiological protection organizations, as well as clarify legal aspects, in which their intervention was framed. The legal framework (Nuclear Law and Regulation ordinances) facilitates the recognition and acceptance, by all involved institutions, of the authority and leadership of the ARN in the application of the radiological protection actions. To this objective, all levels of the involved organizations were encouraged to perform and accept the different roles assigned.

The nuclear emergency exercises carried out during the period 2010-2012 were the following:

- *Villa del Dique and Villa Rumipal, CNE, Exercise 2010, November 18, 2010.*
- *Lima, CNA I, Exercise 2011, November 10, 2011.*
- *La Cruz, CNE, Exercise 2012, September 6, 2012.*

It's been performed a "Public Communication Workshop during Nuclear Emergencies Situations", altogether with the Licensee at Embalse, on September 15 and 16, 2010. The workshop's objective was unifying public and media information criteria during a nuclear emergency. The personnel of the areas of communication and emergency belonging to both CNE and ARN participated of the workshop.

Besides both Licensee and ARN, the following organizations have actively participated and received the specific training according to what the legislation and the Nuclear Emergency Plans establishes:

- National Army.
- National Marine.
- National Gendarmerie.
- Naval Prefecture Argentina.
- Police of Buenos Aires.
- Police of Córdoba.
- Federal Emergency System.
- Civil Defence, Córdoba Province.
- Civil Defence, Buenos Aires Province.
- Civil Defence, Zárate and sub delegation Lima.
- Civil Defence, La Cruz.
- Civil Defence, Villa del Dique.
- Civil Defence, Villa Rumipal.

- Civil Defence, Embalse.
- Embalse Fire-fighter Brigade.
- Lima Fire-fighter Brigade.
- Zárate Fire-fighter Brigade.
- FM radio station El Sitio, Lima.
- FM radio station Libre, Lima.
- FM radio station Delta, Embalse.
- FM radio station Show, Embalse.
- *FM radio station Mediterránea, Embalse.*
- *FM radio station del Lago, Villa del Dique.*
- *FM radio station Neutro, La Cruz.*

All the exercises are carried out with the participation of local people (up to 10 km around the NPP), where the protection actions are practiced. The population's preparation is a key task to minimize the consequences of a nuclear incident. An important part in the preparation is the diffusion of information, so that, everybody will know what to do in case of receiving an alert of a nuclear emergency.

Therefore, previous to each exercise, diffusion activities and training are carried out with special emphasis on primary and secondary schools of the area. The workshops at schools are divided taking into account the differences in the levels of understanding, considering the ages of the students. For this purpose the following four levels are defined:

- Level 1 for children from 6 to 9 years old.
- Level 2 for children from 9 to 12 years old.
- Level 3 for teenager from 12 to 15 years old.
- Level 4 for teenager from 16 to 18 years old.

These activities are carried out once a year during the exercise preparation and its main objective is the diffusion of the protection actions. Similarly, diffusion activities on protection actions are carried out with other sectors of the general public, with talk shows, discussions and explanation / clarification of doubts generated during the open meetings *with the community* organized by the local civil defence, the Licensee and ARN.

All the protection actions diffused at preparation stage are put into practice on an effective way during the Emergency Plan Exercise Application. In this way, all school levels students make the taking cover at local schools and ingest the Iodine pill (simulated with candies). National Gendarmerie, is the organization in charge of the iodine pills distribution to the Emergency Zone population, also performed this function during the exercise. Likewise, all the organizations involved in the exercise tested the efficiency of their functions during this event.

During the Emergency Plan Exercise Application, it was continued implementing a Public Information Centre (CIP) as a potential scenario in case of an emergency. The CIP was put in practice during the last exercise. At the Exercise development, during Green Alert, a statement with the first protection action is issued and then a press conference is given. Then, after the Red Alarm other statements about the situation state are given.

The goal of both conferences is to communicate the state of affairs to the public through a press release and answer questions from reporters. Although it was the first time that this scenario was implemented, the objectives were very well met.

4,700 inhabitants participated in the exercise of CNE 2010 at Villa Rumipal and Villa del Dique. Approximately 10,800 inhabitants participated in Exercise of CNA I 2011 at Lima and 2,000 inhabitants participated in Exercise of CNE 2012 at La Cruz.

During all the emergency exercises, the ARN objective to conduct the emergency response teams belonging to all the civil and law enforcement organizations was achieved appropriately.

Furthermore, the ARN staffs has acquired the necessary skills to carry out the role of conducting civil and Law Enforcement organizations, taking advantage of the operative experience of these organizations. It was also possible to integrate representatives of these organizations to the ECC to conduct the emergency. Notwithstanding, the coordination function assigned to the ARN, each participating organization exercised its own role and competence.

Besides, as were mentioned in section 3.16.2.9. as an implementation of the lessons learned from the Fukushima accident, emergency exercises comprising the long term are being planned.

CNA I:

Every year CNA I staff participate in the internal nuclear emergency exercise, going to the meeting points, and practicing emergency primordial actions. Furthermore, practice of audible alarms within the site is being performed, as well as the CICE conformation, communications with all the Response Forces, the formation of internal response groups, the performance of the Emergency Brigade, coordination with the meeting points and internal distribution of IK pills.

Every two years, exercises are performed of emergency plan implementation with external scope, where the participants are the External Organizations who also participate on the Response Force.

Previously, a Retraining Day is performed with the response organizations: ARN, Zárate Municipal and Province of Buenos Aires Civil Defence, National Gendarmerie, Argentina-Puerto Zárate Coast Guard, Argentina-Navy Naval Base Zárate, Zárate-Campana Police Department, Lima and Zárate Volunteer Firemen, Zarate Naval Base, Radio Lima FM.

Throughout the calendar year, there is a Training Day regarding countermeasures issues in case of nuclear emergency in all schools that are within a 10 km radius from the plant. At the end of each exercise, open discussions are performed in order to identify shortcomings. Such flaws are reported to the ARN through the Report of Exercise Emergency Plan Application. In turn, the CNA I plans improvement areas that are incorporated into the tracking software of pending actions (iWeb), to introduce the corresponding improvements in the following exercises.

CNE:

In CNE NPP, every year the Applicative Exercise of the Emergency Planning is performed. It has validity and effect within the installation and surrounding vicinity within 10 km and defines the methodology, organization and the necessary resources to deal with emergencies that may occur in the plant, such as large fires, major industrial accident, a natural or man induced disaster, an environmental contingency, a nuclear emergency or a combination of these events, and demonstrates how to deal with an accident or emergency situations with prompt actions to nullify or mitigate them.

It involves the entire plant personnel, who play the roles assigned in the event of an emergency, going to the corresponding points or places designated for each particular role, exercising the main emergency actions.

Alarms are issued announcing a state of emergency within the plant, as well as the actuation of the CICE, the application of automatic countermeasures, the practice of Civil Defence Advisory, internal communications and communications with the ARN, with local institutions involved in the emergency, response forces, press releases, etc.

Furthermore, the performance of internal group's response is exercised such as the Fire Fighting Brigade, Medical Service, Physical Safety, Radiation Protection, Dosimetry, Civil Defence Advisors, and assesses the behaviour of external institutions such as police and the National Gendarmerie, whose mission I,s among others, the access control or the distribution of IK pills.

Every two years there is an Applicative Exercise of External Emergency Planning involving, one at a time, the localities that are included within the radius of 10 km, Embalse, La Cruz, Villa del Dique and Villa Rumipal, by intervening all external emergency response organizations. These exercises are coordinated by the ARN, who previously performed retraining, usually joined by CNE staff, of organizations such as Educational Institutions, Neighbourhood Centres, Municipal Civil Defence, Police, National Gendarmerie, News Media, etc. After each exercise, the large amount of information collected is used to provide improved conditions to Emergency Plan through a Monitoring of Corrective Actions resulting from recommendations arisen from the emergency exercises.

3.16.3. INFORMATION OF THE PUBLIC AND NEIGHBOURING PREPAREDNESS

3.16.3.1. AGREEMENTS WITH ORGANIZATIONS CONCERNING EMERGENCY PLANS

The basic aspects mentioned in Section 3.16.2.1 were established not only for the operating NPP's internal and external emergency plans, but also in the agreements with intervening organizations. These agreements, which are part of the emergency plans, enable the use of networks, communication

systems and equipment that belong to organizations, as well as fast transportation, aerial monitoring remote system and fire fighting resources.

With regards to agreements with other organizations, an important example is the treatment of injured, contaminated or irradiated persons *performed in* hospitals. The primary treatment for these cases is foreseen to take place at hospitals located in districts near the NPPs. If specialized health assistance of patients is required, there are agreements with medical institutions, hospital infrastructure and well-trained staff for such cases.

3.16.3.2. BILATERAL AGREEMENT AND THE 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 neighbour country with NPPs.

Argentina has signed agreements related to nuclear issues with Paraguay, Bolivia, Uruguay and Chile. These agreements facilitate bilateral communications in case of trans-boundary release.

Argentina, Chile, Bolivia, Uruguay and Brazil are Contracting Parties of the Convention on Early Notification of a Nuclear Accident, framework to facilitate the exchange of information in the event of a radiological emergency due to a nuclear accident. In the framework of these agreements, representatives of Brazil and Uruguay took part as observers in some of the exercises carried out in Argentina.

3.16.3.3. STATUS OF NATIONAL AND INTERNATIONAL AGREEMENTS

The agreements with national organizations such as the Army, Navy, Gendarmerie, Argentine Coast Guard, National Meteorological Service, Ministry of Health and the Federal Police that are used for the response within an area of 10 km and further, are covered in the following plans:

- Municipal Plan for Nuclear Emergencies, La Cruz, Córdoba, approved in September 2001.
- Municipal Plan for Nuclear Emergencies, Zárate, Buenos Aires, approved in January 2002.
- National Plan for Nuclear Emergencies, Federal System of Emergencies, approved in December 2002.
- Municipal Plan for Nuclear Emergencies, Embalse, Córdoba, approved in June 2003.
- Provincial Plan of Nuclear Emergency, Ministry of Security, Córdoba, approved in November 2003.
- Plan of Nuclear Emergencies, External Aspects, CNE (updated in November 2003).
- Municipal Plan for Nuclear Emergencies, Villa Rumipal, Córdoba, approved in May 2005.
- Plan of Nuclear Emergencies, External Aspects, CNA I (*in update process for future incorporation of CNA II on site*).

In the ARN quality management system, improvements to internal procedures related to emergency preparedness are being continuously performed. Formal and operative activities that have arisen from the Convention of Early Notification and Assistance in cases of Radiological Emergencies for Accidents in Nuclear Centres were met normally and responded to. Additionally, Argentina participates in emergency events and in the CONVEX International exercises.

In connection with the NPPs' safety, emergency preparedness activities have continued. The activities have been focused on disseminating information of the radiological risk and the nuclear emergency plans to the population surrounding the NPPs.

3.16.4. COMPLIANCE WITH THE OBLIGATIONS IMPOSED BY THE CONVENTION

From the preceding considerations, it may be concluded that in Argentina there are updated plans capable of facing emergencies in nuclear installations, and moreover, such plans establish the actions to be followed not only inside the installations but also outside them, and periodic exercises are carried out for their application.

In conclusion, the country complies with the obligations imposed in Article 16 of the Convention on Nuclear Safety.

3.17. ARTICLE 17: SITING

Each Contracting Party shall take the appropriate steps to ensure that appropriate procedures are established and implemented:

- i. for evaluating all relevant site-related factors likely to affect the safety of a nuclear installation for its projected lifetime;*
- ii. for evaluating the likely safety impact of a proposed nuclear installation on individuals, society and the environment;*
- iii. for re-evaluating as necessary all relevant factors referred to in sub-paragraphs (i) and (ii) so as to ensure the continued safety acceptability of the nuclear installation;*
- iv. for consulting Contracting Parties in the vicinity of a proposed nuclear installation, insofar as they are likely to be affected by that installation and, upon request providing the necessary information to such Contracting Parties, in order to enable them to evaluate and make their own assessment of the likely safety impact on their own territory of the nuclear installation.*

3.17.1. INTRODUCTION

The objective of the siting studies is to select a suitable site for a NPP, including appropriate assessment and definition of the related design bases, taking into account that NPP's design implies the consideration of site dependant factors which may affect, directly or indirectly, the plant safety. For instance, the capability and reliability of the ultimate heat sink and power supply networks, the potential occurrence of natural and/or man induced events, and the characteristics of communication routes and accesses.

Therefore, the siting studies aim at determining the effects of external events occurring in the region of the site, to evaluate the potential radiological impact on the environment due to the plant operation and the feasibility of the emergency plans.

In Argentina, the studies (for selecting the location of a NPP) are part of the requirements the Licensee shall comply with at the time they request a construction license, or included in the Periodic Safety Review (PSR), document necessary to require the Operating License renewal. A previous and independent licensing of a site is not explicitly required.

The results of siting studies of the NPPs were used in determining parameters required for the application of models describing radionuclide dispersion to the environment. These models enabled the evaluation of dose exposure due to radioactive effluents released during normal operation.

Moreover, the information supplied by siting studies enabled to foresee the implementation of actions required to protect the public from accidental situations. These actions were taken into account in the elaboration of the corresponding Emergency Plans.

3.17.2. EVALUATION OF SITE RELATED FACTORS

3.17.2.1. NORMATIVE ASPECTS

In Argentina, a NPP construction must not be initiated without a previous Construction License issued by the Regulatory Body, upon request from the Licensee.

In line with this approach, at the time of applying for the Construction License, the Licensee must submit to the Regulatory Body all the documentation required to evaluate the radiological and nuclear safety of the installation to be built, including the site characteristics in relation to:

- Natural and man-induced external events that could affect the installation safety.
- Dispersion of radionuclides to the environment, both in normal and accidental conditions.

The Regulatory Body issues the license once the Licensee has demonstrated that the design of the NPP to be built complies with the regulatory standards and other specific regulatory requirements for the selected site, taking into account the NPP-site interaction.

Besides, the preliminary version of the new Regulatory Standard AR 10.10.1. "Siting of Class I Nuclear Installations" and the corresponding two Guides are being developed and will take into account the lessons learned from the Fukushima accident as long as those lessons be incorporated in the 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 Río Tercero Dam Lake, in the Province of Córdoba.

At Atucha site there are two independent units, CNA I in operation, and CNA II under construction. At Río Tercero site, CNE is in operation.

The site studies performed for CNE and CNA I in each case underwent 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.

3.17.2.3.2. Siting studies under execution

The CAREM-25 prototype reactor is being constructed 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 Nucleoeléctrica Argentina S.A. NA-SA, duly extrapolated and updated were made available to CNEA (National Atomic Energy Commission of Republic Argentina), responsible for the design, construction and operation of the CAREM prototype reactor. These studies include, inter alia, external hazards, population density & distribution, and NPP lay out.

These studies are 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 ([see section 3.17.2.3.3.1.](#)).

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.

Recently the siting studies of the CAREM reactor have been concluded and, CNEA has applied for the construction authorization which is being evaluated by the Regulatory Body.

3.17.2.3.3. Site re-evaluation

The selection of CNA I, CNA II and CNE sites, were performed, according to the criteria and the information available, during the sixties and the early seventies.

Nowadays, there are more accurate tools to define the design parameters than those existing at that time. Therefore, among others, the following reasons make necessary to re-evaluate these parameters including an analysis of its impact on the current safety condition:

- Existence of new and more reliable data and information for assessing the occurrence of external event hazards, in comparison with those available at the time of the original design.
- Availability of Methods and criteria consolidated through their use in different regions (i.e. different natural 'environments') and countries (i.e. different regulatory and practice 'environments') allowing to reach certain consensus in the international practice about what to do and how to do it.

- A more balanced situation between the regulatory requirements and the industry practices, through the use of more realistic and integrated criteria, with a trend to reduce the use of excessive conservatism.
- Changes in both the real plant configuration and the present conditions in the nearby region; as well as those modifications introduced to structures, systems and equipment to improve their performance.

The design parameters corresponding to earthquakes, extreme meteorological phenomena and man induced events were determined according to the region and site specific conditions of each plant.

The Argentine's NPPs seismic design is consistent with the national regulation, AR 3.10.1. "Earthquake Protection in Nuclear Power Plants", and international criteria established at the NPP's 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 with the purpose of apply the corresponding lessons learned, the Regulatory Body requested perform a stress test to each Argentinean NPP (see Annex VIII) 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 were showed weaknesses or critical situations (cliff edge effects) which make necessary to take urgent actions.

The improvements and modifications proposed by the NPPs Licensees included an implementation schedule composed by short, medium and long term actions; the latter reaching the year 2016. Finally, as result of the stress tests, it was issued a regulatory requirement containing numerous improvements and modifications to be implemented by the Argentinean NPPs, some of which have already been implemented.

Below there is a summary of the re-evaluation of siting aspects at different times after the original evaluation, as well as those being carried out at present.

3.17.2.3.3.1. CNA I and CNA II site re-evaluation

For Atucha siting, the following re-evaluations were performed:

- Re-evaluation of tornadoes and severe storms hazard, including the energy transmission lines which are essential for the plant safety. A working plan was prepared, starting in December 1998. A complete database has been compiled adding the 20 years of additional data, since 1980, and a more refined model for assessing the tornado impact probability was used (i.e. Twisdale and Dunn).

- 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 undergo continuous training.
- Different alternatives have been studied regarding flooding in CNA I. 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.
- 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 will improve safety for these scenarios providing an independent system from the actual residual heat removal system, which uses the river as the final heat sink.

The analysis and assessments, as part of the stress tests carried out in 2011 / 2012, required by the Regulatory Body related with extreme external hazards (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:

3.17.2.3.3.1.1. Earthquakes

Atucha site is located in a region of low seismicity. The SSA based on the corresponding updated seismic hazard is being carried out since 2011 and the objective is to demonstrate the seismic safety margin for CNA I and CNA II for the RLE and to confirm the non-existence of cliff edge effects.

The updated seismic hazard is based on the comparison of two Probabilistic Safety Hazard Analysis (PSHA) performed by AECL and JJJ & Associates on behalf of the Licensee, which follows the principles outlined in the IAEA Safety Guide SSG-9 "Seismic Hazards in Site Evaluation for Nuclear Installations".

The maximum ground acceleration (PGA) for Atucha site was estimated in 0.1 g., being adopted as RLE having a sufficient margin over the original DBE in order to improve the safety and find weaknesses that could limit their capacity to support the consequences caused by an earthquake greater than the DBE. Based on the PSHA outcomes, the RLE adopted for SSA of CNA I and CNA II is derived from the mean uniform hazard response spectra (UHRS) for a recurrence period of 10,000 years.

The CNA I NPP was not originally designed or qualified considering severe earthquakes. However, due to the conservative design applied as well as the SSCs robustness, it was considered that there is an inherent capability to withstand earthquakes of a certain level which will be determined by means of a SMA to assess the SSCs' status in relation to their ability to perform its safety function after a specific earthquake occurrence.

The CNA II original design criteria were based on a 0.05 g PGA DBE. Additionally, design principles and construction measures for low seismicity regions were applied.

Considering the above mentioned, 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.

New hydrologic and hydraulic study involving Atucha site is in process (see section 3.17.2.3.3.1.2) which considers the broken dams located upstream to Atucha site due to an earthquake exceeding the dam design basis that cause flooding beyond the plant flooding design basis. Besides, during the phase 5 of the SSA is foreseen to be analysed an earthquake exceeding the DBE that may cause internal flooding or low water level beyond the design basis. These analyses were scheduled to be completed by the end of 2015.

CNA I

The CNA I specific SSA consists of five phases, namely:

- Phase 1: Scope of the study and preliminary plant inspection walkdown.
- Phase 2: Development of the Safe Shutdown Equipment List (SSEL) and Systems Walkdown.
- Phase 3: Seismic response and in structure response spectra determination.
- Phase 4: Inspection Walkdown of the seismic capacity and screening process.
- Phase 5: Structure capacity evaluation and detailed analysis.

Phase 1 was completed in December 2011. Phases 2 to 4 were also completed in May 2013. From those components that were not screen out some are easy fixes that were solved during 2013 scheduled outage, some need a circuit and relay analysis and others HCLP and anchorage capacity calculation. A few will be evaluated during phase 5, which is foreseen to be performed by 2015.

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 plans to add a pump (fourth pump) to its secured cooling water river system (UK) which will be 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:

Easy fixes:

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. The following easy fixes were already implemented:

- Electrical and I&C cabinets in raised floor: an anchor was be 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.

Seismic instrumentation:

The installation of new seismic instrumentation consisting in a sensor located in the free field, common to CNA I and CNA II, plus 4 accelerometers within each plant following NRC-RG-1.12 and IAEA recommendations with alarm/announcement in MCR. Implementation is scheduled by 2014.

Seismically induced internal flooding:

The possible seismically induced internal flooding was analysed during the systems walk downs.

CNA II

The SSA consists of the same five phases showed for the CNA I case. Phase 1, was carried out in October 2011. Phases 2 to 4 were completed in June 2013. The phase 5 is foreseen to be developed before beginning the nuclear commissioning.

As a result of the mentioned assessments, it was decided to implement the following improvements:

Plant walkdown:

A plant walkdown to verify the seismic capability was finished in June 2013, resulting that all the outliers in the SSEL; which consider issues like systems interaction, construction and installation, maintenance and housekeeping, as well as the relevant modifications to be carried out; were identified. All easy fixes found will be implemented during 2013.

The results of the stress tests generated by the NPP's Licensees 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 was confirmed the non-existence of cliff edge effects.
- 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.
- It has been analysed internal and external flooding caused by earthquakes and 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 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.

3.17.2.3.3.1.2. Flooding / low water level

The estimated maximum water height that would be reached on the Atucha site after the rupture of the Yaciretá 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. 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 cooling system (UC) pumps and the assured cooling system (UK) pumps are located at a level of 6 meters and may be vulnerable to river level rise due to it is less than the PMH of 8.45 meters. Besides, the minimum water level height for which the plant can operate is -1 meter given by the UK pumps water intake level.

To overcome these CNA I extreme high and low water levels design weaknesses, it is being installed a fourth pump UK in a CNA II pumps house. This pump will keep running even with a river level rise of 8.45 meters or low river levels of -2.00 meters. Therefore, this will allow 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 assured 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.

Related 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 be affected by the Paraná river's high or low levels. From the functional point of view, considering the UK system loss, the pools will 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 proposed to be implemented by the end of 2013.

New hydrologic and hydraulic study, in order to supplement, update, and also make a prospecting considering future possible scenarios is in process. It is foreseen 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. Besides, this study considers the broken dams located upstream as well as the mouth Paraná River boundary condition given by the Río de la Plata river levels. This study was scheduled by the end of 2015.

During the phase 5 of the SSA is foreseen to be analysed an earthquake exceeding the DBE that cause flooding or low water levels beyond the flooding design basis or the low water level design basis. This improvement is foreseen to be implemented also by the end of 2015.

Besides, the Licensee has decided to carry out the following improvements:

CNA I:

- Based on studies of boundary conditions of operation for decreases in the river levels values, a procedure for systematizing the outage of the plant in case of extreme low-water-levels is being developed.
- Installation of an additional (fourth) pump to the river Water Cooling Ensured System (UK) capable of withstanding the maximum level given by the breaking of the Yaciretá dam upstream. Implementation is scheduled by the end of 2014.
- Increase in time water inventory reposition to the SGs independent of secured river water by the end of 2014.

CNA II:

- Procedures for cooling the plant via the SGs in case of SBO and loss of the assured service water cooling system PE due to flooding or low water level (as a result of the analysis performed in the PSA) are in progress. This improvement is foreseen to be implemented by the end of 2013.

The Regulatory Body carried out the assessment of the stress tests results and concluded the following:

- The need for some regulatory actions but not relevant weaknesses that require urgent actions.
- It has been verified the Licensee complies with both the design and licensing basis.
- For the purpose of increasing the capacity to respond to extreme situations the Licensee proposes to implement a set of improvements including the corresponding implementation schedules, which were considered acceptable.
- The consideration of flooding/low-water-level for Argentine NPPs is consistent with both domestic regulations and international criteria established at the design time. However, further studies were considered necessary to require for the Atucha site. A new hydrologic and hydraulic study is in progress including a review of background studies, and to carry out a survey considering possible future scenarios.
- Internal and external flooding situations have been analysed. Regarding this issue is considered that the Licensee is carrying out appropriate actions to successfully cope with these scenarios.

3.17.2.3.3.1.3. Other external hazards: tornadoes; wind loads; lightning and intense rains

Other possible external sources of flooding different from Parana 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 object impacts thrown by tornadoes affecting the plant buildings. This re-evaluation was scheduled by the end of 2013.

Besides, a re-evaluation of the risk of tornadoes for the Atucha site is foreseen to be completed by the end of 2015. This re-evaluation will include the response analysis and the existing margin for the safety related buildings and SSCs facing BDB tornadoes.

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 was decided to implement additional studies to confirm them.

Based on the assessments carried out, the Regulatory Body concluded that:

- The needs for some regulatory actions but not relevant weaknesses that require urgent were identified.*
- It has been verified that the Licensee complies with both the design and licensing basis.*
- The consideration of tornadoes and wind loads for Argentine NPPs is consistent with both domestic regulations and international criteria, established at the design time. However, it was considered necessary to require new studies re-evaluating the risk of tornadoes for the Atucha site, which are in process. The result of this re-evaluation will determine if it is necessary to require complementary actions.*
- Tornadoes, wind loads, lightning and intense rains have been analysed and it is considered that the licensee is carrying out the appropriate actions to successfully cope these scenarios.*

3.17.2.3.3.1.4. Regulatory Actions

As 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 improvements and modifications referred to seismicity, flooding and other extreme external hazards to be implemented in each NPP. This regulatory requirement includes short, medium and long term actions; the latter reaching the year 2015.

3.17.2.3.3.2. CNE site re-evaluation

A list of external hazards for re-evaluation was prepared and they were prioritised in accordance with its safety impact on the original design, as follows:

- Earthquakes.
- Extreme meteorological phenomena (tornadoes and severe storms).
- Man-induced events (mainly, explosions and fires, external to the plant site).

The following actions were carried out related to each external hazard considered:

- Collection and analysis of related documents and reports.
- Regarding the re-evaluation of the operational response in case of an earthquake occurrence, two regulatory requirements were issued in 1999, regarding the implementation of an updated program of plant response to an earthquake occurrence. These requirements include a re-evaluation of the seismic safety within the framework of an integrated, systematic and updated program. Therefore, the plant accomplished the definition, procurement, installation and commissioning of new digital seismic instrumentation to detect seismic events, evaluating their severity and providing data to plant operators. The instrumentation, installed during the 2000 planned outage, allows the recording of the seismic activity and provides this information directly to the operator in the control room for decision making in case of the occurrence of a seismic event.
- Preparation and implementation of operating procedures (update and improvements) to assess plant physical damage and plant operational situation after an earthquake occurrence, and, thus, to help in the decision making process for continuing operation and long term plant safety assessment. It includes a number of inspections to be carried out to determine the status of the safety system and safety related systems and according to the inspection results, to determine the full power operation, hot shutdown, cold shutdown or plant start-up.
- Re-evaluation of tornadoes and severe storms hazard, including the energy transmission lines which are essential for the plant safety. A work plan was prepared, starting in December 1998. A complete database has been compiled adding the 20 years of additional data, since 1980, and a more refined model for assessing the tornado impact probability was used (i.e. Twisdale and Dunn).
- Concerning the re-evaluation of potential hazards from man-induced events at the plant site, the document of Design Basis Threat is periodically reviewed by NA-SA, and experts from the national organization in charge of the physical protection of the NPPs are continuously trained.

A revision of the risk analysis for external events and the assessment of the radiological environmental impact are going to be performed as a part of the CNE Periodic Safety Review. In particular, as was mentioned, a seismic re-evaluation is planned as part of the CNE Life Extension Project.

The analysis and assessments, as part of the stress tests carried out in 2011 / 2012, required by the Regulatory Body related with extreme external events (earthquakes; flooding / low level water; and others external events) were performed. The improvements proposed by the Licensee as the stress tests results include the following:

3.17.2.3.3.2.1. 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, preliminary Embalse specific Uniform Hazard Spectra (UHS) corresponding to a non-exceedance probability level of 0.0001 per annum were developed in 2008, which reflect the knowledge at that time.

Geological investigations have since proceeded in order to fully characterize the seismotectonic setting of the Embalse region and quantify local fault activity. These investigations include the following activities *which have been completed by the end of 2010:*

- 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.

Recently NA-SA decided to perform a comprehensive review of the site seismic hazard to be used in the SSA through a SMA based on a PSA which is being performed. The SMA will determine the existent seismic margin to resist earthquakes bigger than DBE and it is expected to be finished during the plant refurbishment. The SMA will provide a measure of the plant's robustness to face a RLE (HCLPF).

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 higher than 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 caused by a seismic event greater than the DBE.

As a result of the mentioned assessment, it was decided to implement the following improvements:

Operating procedure

An operating procedure supported by the SMS has already been developed, which list all the foreseen activities to be implemented 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.

Emergency Core Cooling System (ECCS)

Based on the SMA walkdown and fragility analysis findings, the seismic capacity of the medium (MP) and low pressure (LP) ECCS will be improve. It is foreseen to be implemented by the end of 2015.

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

A 24-inch rupture disc (RD) assembly shall be installed on the top of the existing calandria vault inspection port to provide additional pressure relief to maintain the calandria vault integrity following a severe core damage accident. It is foreseen to be implemented by the end of 2015.

Spent fuel dry storage system (ASECQ):

The silos 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 silos 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.

The results of the stress tests were assessed by the Regulatory Body and like in the Atucha site case; it agrees with the approach determined by the Licensee for the SSA re-evaluation and consider that:

- The evaluation of safety margins is consistent with the defence in depth concept.
- It has been analysed internal and external flooding caused by earthquakes and is considered that the Licensee is carrying out the appropriate actions to successfully meet these scenarios.
- For the purpose of increasing the capacity to respond against extreme conditions, the Licensee proposes to implement a set of improvements that are acceptable.

The Regulatory Body with IDIA as technical support organization (TSO) has recently concluded a site seismic hazard review of the assessment conducted by the Licensee. The results allowed the Regulatory Body to define the following regulatory framework for CNE life extension project:

- New SSC's designs must be done considering, as design basis, the load obtained from the updated UHS and;
- In relation to SMA, it was stated that too verify the reactivity control and core heat removal safety functions, a PGA of 0.39 g must be adopted as RLE. For the associated SSCs the acceptance criteria is $HCLPF > 0.39 g$.

3.17.2.3.3.2.2. Flooding / low water level

The CNE is located at the Embalse site in the Cordoba 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 correspond 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 intake from the lake is located far away from the shore in the Rio Tercero riverbed level which is the main tributary to the lake. Historical background 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 case of a dam break occurrence caused by an earthquake higher than the RLE.

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 coolant pumps, low pressure emergency core cooling pumps, compressed air for instruments and some coolers. It was estimated that in case the main coolant pumps go out of service, the primary heat transport system (PHTS) is sufficiently cooled because the heat to be extracted would be extremely low. 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:

- Re-evaluation of the consequences of probable earthquakes on the existing dam located downstream of CNE. It is considered that this analysis will determine if it is necessary to implement additional improvements related to a very low lake water-level. This re-evaluation is planned to be finished during the plant refurbishment and completed by 2015.
- An air compressor driven by a DG located in the turbine building has being installed in a higher level (100 meters). This is connected by valves to the air supply tanks of instruments which are necessary for the operation actions required to cooling the reactor.
- The secondary control room gateway located in the turbine building shall be reinforced. This improvement is expected to be implemented during the plant refurbishment and completed by the end of 2015.
- Relocate the site where is located the centre of the plant lighting system to avoid the internal flooding that could be caused by an eventual water leak from the condenser. This improvement is foreseen to be implemented during the plant refurbishment and completed by the end of 2015.

The Regulatory Body carried out the assessment of the stress tests results and concluded the following:

- The need for some regulatory actions but not relevant weaknesses that require urgent actions.
- It has been verified the Licensee complies with both the design and licensing basis.
- For the purpose of increasing the capacity to respond to extreme situations the Licensee proposes to implement a set of improvements including the corresponding implementation schedules, which were considered acceptable.
- The consideration of flooding/low-water-level for Argentine NPPs is consistent with both domestic regulations and international criteria established at the design time. However, further studies were considered necessary to require for the Embalse site. For CNE, is foreseen to carry out a re-evaluation of the consequences of the occurrence of earthquakes on the existing dam located downstream of the plant.
- Internal and external flooding situations have been analysed. Regarding this issue is considered that the Licensee is carrying out appropriate actions to successfully cope with these scenarios.

3.17.2.3.3.2.3. 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 is foreseen to be completed by the end of 2015. This re-evaluation will include the response analysis and the existing margin for the safety related buildings and SSCs facing BDB tornadoes.

A recent plant outage caused by loose material accumulation in the plant water intake occurred revealing a weakness existence. This loose material came from extreme drought and ground removal which affected large areas surrounding the plant as well as the action of intense rains and strong winds which swept this material. The measures taken as a consequence of this event are:

- *Hiring divers to make periodic cleaning of the plant water inlet.*
- *Take steps to restrict planting and grazing livestock on land adjacent to CNE as well as the weeding or clearing of these areas.*

Based on the assessments carried out, the Regulatory Body concluded that:

- *The needs for some regulatory actions but not relevant weaknesses that require urgent were identified.*
- *It has been verified that the Licensee complies with both the design and licensing basis.*
- *The consideration of tornadoes and wind loads for Argentine NPPs is consistent with both domestic regulations and international criteria, established at the design time. However, it was considered necessary to require new studies re-evaluating the risk of tornadoes for the site, which are in process. The result of this re-evaluation will determine if it is necessary to require complementary actions.*
- *Tornadoes, wind loads, lightning and intense rains have been analysed and it is considered that the licensee is carrying out the appropriate actions to successfully cope these scenarios.*

3.17.2.3.3.2.4. Regulatory Actions

As 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 improvements and modifications referred to seismicity, flooding / low water level and other extreme external hazards to be implemented in each NPP. This regulatory requirement includes short, medium and long term actions; the latter reaching the year 2015.

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

The authorized limits to the environmental releases of the Argentinean NPPs under operation were set by the ARN for relevant radionuclides. The radiological protection criteria used by ARN to control the dose received by workers are consistent with the latest ICRP recommendations (see article 3.15.). The authorized limits to the environmental releases have not been modified respect than were reported in the fifth Argentinean National Report to the CNS.

The Argentinean NPPs operator disposes of Operating Procedures for Abnormal Events (POEAs) foreseen to manage abnormal events and mitigate the corresponding consequences. The following new POEAs are being issued as a result of the assessment developed in the stress tests:

- *Increase significantly in time the water reposition to the SG for several beyond design basis scenarios. (CNA I).*
- *Instruction for the loss of a 24 V rectifier (CNA I).*
- *Increase up to 7 days the reposition of water to the dousing and the SGs, is under an independent revision stage (CNE).*
- *Instructions for maintaining spent fuel cooling for several beyond design basis scenarios. (CNA I & CNE).*

A Severe Accident Management Program (SAMP) considering severe accidental situations continue being developed for CNA I and CNE and, some strategies have been identified as part of the L2 PSA evaluations for the CNA II case (see article 3.14.). The following Severe Accident Management Guidelines have been developed:

- *Instructions related to the modification of the mandatory documentation of the Operation Manual, to be applied in the C.I.C.E (Internal Centre of Emergency Control) and the in the main control room (CNA I).*
- *In the case of CNE the project consists of four sets of guidelines (which are currently being re-evaluated by the Licensee):*
 - *SACRGs: Severe Accident Control Room Guidelines.*
 - *SAGs: Severe Accident Guidelines.*
 - *SCGs: Severe Challenge Guidelines.*
 - *SAEGs: Severe Accident Exit Guidelines.*

In case of a radiological emergency, the actions to be taken inside and outside the NPPs, by all the intervening organizations, are considered in the application of each Emergency Plan (see article 3.16.). Emergency Plans were updated after the analysis and assessments developed as part of the stress tests carried out in 2011 / 2012, related with emergencies management and control, fuel damage mitigation, radioactive emissions reduction, revision of procedures, personnel training and equipment availability).

The Environmental Management System (EMS) implemented by the Licensee adopts the requirements of ISO 14001:2004 "Environmental Management Systems-Requirements with guidance for use". The Management Manual and the procedures that complement it, provides the corresponding information and describes the EMS.

The Licensee Environmental Policy is developed, implemented, conducted, reviewed and maintained through the EMS. The EMS is applicable to the activities, processes, products and services that interact with the environment and that NA-SA may control and over which it has influence. The EMS scope includes:

- Power generation in CNA I.*
- Power generation and Cobalt 60 production in CNE.*
- Maintenance, construction and commissioning of CNA II.*
- Activities related to the Project of Life Extension of CNE.*
- Activities related to Fourth NPP 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.*
- Communicating 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.*
- To promote the training of personnel in the care of the environment.*
- Continuously improve the environmental performance.*

3.17.3.1. ENVIRONMENTAL RADIATION MONITORING

The Licensee performs the environmental radiological monitoring around the Argentinean NPPs. Activity concentration measurements are performed in different environmental matrices as well as environmental dose rate measurements in air. The results are compared to the values sets in national and international standards, as well as to the values obtained from environmental dispersion models to control the radiation protection of the population. It should be noted that the environmental radiological monitoring conducted by the Licensee is completely independent from that performed by ARN. Additionally, this task allows answering the concerns of public opinion on the subject.

3.17.3.1.1. Atucha site

In this section is described a summary of the radiological environmental monitoring around CNA I carried out by NA-SA and ARN.

The location of the radiological sampling sites is around CNA I, mainly within a radius of 10 km centred on the installation.

Samples are taken, representative of the various compartments of the matrix environmental of radionuclide transfer. To assess the environmental impact of liquid discharges, from different water bodies, sediments and fishes, samples are taken and analysed. Additionally, monitoring is performed of drinking water from wells near the plant. To assess the environmental impact of discharges to the atmosphere, samples of air, soil and food produced in the zone (such as milk and vegetables) are taken and analysed. Additionally, a measurement of environmental dose rate is performed. The levels of tritium in samples of condensate (moisture) are also determined.

The above mentioned samples of water as well as of in-depth sediments and shore sediments are taken about 3 km downstream and approximately 5.1 km upstream from the plants (background sampling point).

The ground water samples are taken at a school, located 7.1 km from the plants, being the background point. Additionally, some samples were taken at a site located 3 km downstream from the plant. Furthermore, samples of drinking water around the area are taken. In the same areas where samples of river water are taken, also fish samples are taken.

Milk samples are collected weekly from pasture land located within a 5 km radius of the plants.

Regarding air samples, three sampling points are used, located within 10 km from the plant. This monitoring is conducted in the form of total aerosols (TSP, Total Suspended Particulate Matter), where all the aerosols present in an air sample is collected on a high efficiency filter.

Conclusions:

- The measurement results from the various samples analysed indicate that it has been detected the presence of tritium on moisture condensate samples, food and in one sampling point of river water. However, the values are insignificant from the dosimetric viewpoint and are consistent with those obtained through dispersion modelling into the environment.
- In case of drinking water samples, it is noted that all the analysed results of alpha-beta activity proved to be lower than the corresponding values of 0.5 and 1.0 Bq/l, which were established by the World Health Organization (WHO), being unnecessary perform additional analysis of specific radionuclides.
- It has been detected the presence of traces of Strontium 90 in some vegetable samples, being the measured values compatible with the environmental fallout levels reported by different countries. In addition, traces of Cesium 137 detected in some samples of sediments and soils are consistent with the values reported by the international bibliography to sites unrelated to nuclear facilities published by the Scientific Committee of the United Nations for the Study of the Effects of Atomic Radiation (UNSCEAR 2000-2008).
- The average environmental dose rate measured in the area is consistent with the average value reported by UNSCEAR for sites not influenced by nuclear facilities.
- With regard to the measurements of the tritium concentration in water samples, it was verified that these results comply with the limits and dose constraints for members of the public established in the AR 10.1.1. Standard "Basic Radiological Safety Standard", which are consistent with the recommendations of the International Commission on Radiological Protection (ICRP) and the IAEA. In addition to comply national law, it was verified that the radionuclide concentrations results measured in drinking water do not exceed the reference values recommended by the WHO for drinking water intake.
- From the analysis performed of all the activity concentration results measured in the different samples analysed from the environmental monitoring around CNA I, it is concluded that these correspond to those obtained by calculation models to estimate the dose to the most exposed individuals, and are lower than those established in current regulations.

Moreover, the individual average dose corresponding to the critical group of CNA I is well below the design dose constraint, set for a particular installation at 0.3 mSv. This dose was determined from controlled discharges of radioactive effluents to the environment, measured by the NPP's facilities, and the implementation of environmental transfer models recommended internationally (see section 3.15.2.1.).

3.17.3.1.2. Embalse site

In this section is described a summary of the radiological environmental monitoring around CNE carried out by NA-SA and ARN.

The environmental radiological sampling points around CNE are located in a zone defined by a 10 km radius centred at the plant.

As mentioned in the CNA I case, around CNE samples are taken, representatives of the various compartments of the radionuclide transfer environmental matrix.

To assess the environmental impact of liquid discharges, samples from different water bodies, sediment and fish are taken and analysed. Additionally, drinking water monitoring from the Embalse city downtown area is performed.

To assess the environmental impact of discharges to the atmosphere, samples of air and food produced in the area (such as milk and vegetables) are taken and analysed. Additionally, a measurement of the environmental dose rate is performed. The levels of tritium in condensate moisture samples are also determined.

The ground water samples are taken in a sample point located about 2 km from CNE. Tritium concentration are determined monthly in lake water samples, drinking water and groundwater, and the concentrations of Cesium 137, Cobalt 60 and Strontium 90 are quarterly determined.

Conclusions:

- The results of measurements performed in the various samples analysed indicate that it has been detected the presence of tritium on moisture condensate samples, lake water samples, drink water and food. Moreover, traces of Cesium 137 were detected in some fishes, and sediment samples.*
- The presence of tritium in the lake water, in drinking water and in food samples, is due to the controlled discharge of effluents generated in the processes of water purification and degassing of the reactor primary cooling circuit. In addition the tritium concentration level of the lake water is related to seasonal variations in the dam water level.*
- The presence of traces of Cesium 137 in some fishes and sediment samples is attributable to the environmental fallout. The levels observed are similar for samples analysed both upstream and downstream of the plant.*
- Regarding the measurements of the tritium concentration in water samples, it was verified that these results comply with the limits and dose constraints for members of the public established in the AR 10.1.1. Standard "Basic Radiological Safety Standard", which are consistent with the recommendations made by ICRP and IAEA. In addition to comply national law, it was verified that the radionuclide concentrations results measured in drinking water do not exceed the reference values recommended by the WHO for drinking water.*
- The average environmental dose rate measured in the area is consistent with the average value reported by UNSCEAR for sites not influenced by nuclear facilities.*
- From the analysis of all the activity concentration results measurements in the different samples analysed from the environmental monitoring around CNE area, it is concluded that these correspond to those obtained by calculation models to estimate the dose to the most exposed individuals, and are lower than those established in current regulations.*

On the other hand, the individual average dose corresponding to the critical group of CNE is well below the design dose constraint, set for a particular installation at 0.3 mSv. This dose was determined from controlled discharges of radioactive effluents to the environment, measured by the NPP's facilities, and the implementation of environmental transfer models recommended internationally (see section 3.15.3.2.).

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 cover the necessary design and construction aspects in order to prevent accidents as well as to mitigate their radiological consequences if they occur.

These standards are compatible with deterministic concepts such as the defence in depth principle, and incorporate probabilistic concepts in order to define design criteria for the NPPs.

The NPPs design is in accordance with the defence in depth principle and complies, in addition, with the criteria of redundancy, physical separation (segregation) and diversity specified by the Argentine Regulatory Standards.

In addition, requirements taking into account the prevention of SSC's eventual degradation, maintenance of safety systems reliability levels, and implementation of an emergency plan are included in the respective Operating Licenses.

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.

CNA II is at the [commissioning stage](#). It is conceptually similar to CNA I, but it 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 base material of the RPV, easiness for early detection of fissures 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.

The CAREM reactor, a project prototype of a small power NPP, is in an advanced design stage and has applied to obtain the construction authorization. The reactor has 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, those AR standards are already being fulfilled or are being implemented. The AR Standards have been applied for CNA II design, construction and commissioning. Besides, the AR standards have periodically been subjected to a harmonization process against IAEA's standards. Currently, the AR standards are being re-evaluated to incorporate the lessons learned from the Fukushima accident.

The fuel elements are controlled, inspected, tested and verified according to the guidelines established in each installation's QA program, which comprises manufacture, transportation, reception and use stages.

The primary circuit integrity for both normal and design basis accidental conditions is preserved considering the effect of anchorages, connections, internal and external loads and deformations caused by thermal, mechanical and irradiation effects.

The NPPs have a containment representing the last confinement barrier, designed to support loads due to different accidental situations both from internal and external initiating events.

The containment system design criteria specially takes into account the number of confinement barriers, its retention capacity for radioactive material, its behaviour under normal and accidental loads, the leakage rate to the atmosphere and the results of the verification tests.

The shutdown systems (control rod drop and liquid poison injection) design criteria ensure the reactor shutdown in normal and accidental situations, keeping the safe state for the necessary time period.

The Emergency Electric Power Supply System design criteria allow the preservation of an adequate safety margin under normal and accidental conditions. It also complies with the independence, redundancy, physical separation and diversity criteria. External events such as fire and missiles are also considered.

Regarding CAREM reactor, considering that this is a prototype of an innovative design, the ARN has established an ad hoc licensing scheme, applicable to construction and commissioning stages.

3.18.3.1. CNA I NUCLEAR POWER PLANT

The reactor safety systems design and confinement barriers preventing fission product release, such as fuel element claddings, primary circuit and reactor containment, comply with the criteria established in AR Standard 3.2.1. Moreover, the safety systems design complies with the single failure as well as with the segregation and diversity criteria.

The core heat removal system design complies with the requirements of AR Standards 3.3.2. and AR 3.3.3. under normal operation (heat transport primary system and shutdown cooling system) and during hypothetical accidental situations (emergency core cooling system).

Both CNA I shutdown systems design comply with the criteria established in Standard AR 3.4.2., particularly as far as diversity, redundancy and reliability concerns.

The following systems constitute CNA I confinement barriers, as required by the Standard AR 3.4.3.:

- The containment system: this system is constituted by a steel sphere of approximately 50 m in diameter enclosed by a second safety cover of concrete, as its external shield. The system includes several penetrations, air locks and the isolation contention sub-system.
- Radioactive material removal system in case of an accident: this system is located between the steel sphere and the external shield and operates by passing contaminated air through carbon and absolute filters.

CNA I design also complies with the requirements of Standards AR 3.2.1., AR 3.3.1., AR 3.4.1. and AR 3.4.3., particularly regarding the uncertainty data boundary, and the application of safety concepts valid when its design was developed, such as redundancy, diversity, etc.

Such verification was obviously indirect, since the purchase contract specified that the SSCs, designed and manufactured in the Federal Republic of Germany, had to comply with the German standards because CNA I design had to be licensable in that country.

On the other hand, methods and calculation tools compatible with the state-of-the art of those times and verified through operation experience were used in the core design.

Regulatory Standard AR 3.2.1. criteria, related to the operator performance, are also fulfilled. The operator may always make provisions to avoid a situation that could affect the NPP's safety, but he must not avoid the necessary operation of safety systems. In any state of the NPP, all manually executed commands are subordinated to the reactor's protection system; therefore, reactor safety is not threatened by the non-detection of measurement device readings or alarm signals, or any human error that could occur.

The Regulatory Standard AR 3.4.1., concerning man-machine interface taking into account the state-of-the art at the time the NPP was designed, regarding information processing and report systems is fulfilled. Particularly, during an appropriate time interval after the automatic activation of a safety system, no action is required by the operator who, on the other hand, is unable to avoid or interrupt its operation. Nevertheless, the operator may initiate other safety actions.

A new spent fuel element storage facility is being constructed because the available positions in the fuel storage pools will be completed by February 2015 (see section 3.19.9.3.1.).

3.18.3.1.1. Design improvements implementation

As was mentioned in sections 3.14.3.1.1.; 3.14.3.1.2.1.; 3.16.2.9. and 3.17.2.3.3.1.; 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 measures / changes have been foreseen to be implemented to prevent beyond design basis accidents or to mitigate their radiological consequences:

- Light water reposition to primary system through the pressure and inventory control system.*
- Restoration of external power supply.*
- Passive components control for spent fuel storage pool system.*
- New Emergency Power Supply System (EPS).*
- Electrical interconnection between normal bars of CNA I / CNA II.*
- Alternative power sources (MDGs).*
- Extension of the batteries availability. Disconnection of electrical loads.*
- 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).*
- Dose rate's remote measuring system.*
- 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).*
- Alternative cooling mode of the DGs (new EPS).*
- Easy fixes coming from SSA program.*
- Installation of new seismic instrumentation.*
- Installation of an additional (fourth) pump to the river Water Cooling Secured System (UK).*
- Increase in time of water inventory reposition to the SGs.*

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 IV). 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 is to carry out the activities which are required to put CNA II in operation.

On the other hand, the licensing activities for CNA II are carried out by ARN through the regulatory system described in article 3.7. of the present report. The regulatory activities consist principally 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 N° 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 exists no restraint to continue the licensing process of CNA II, as far as the Licensee fulfils the legal regulatory system in force that includes specific requirements that arise from safety evaluations and inspections that will be performed in the future.
2. For the granting of the Commissioning License, the Licensee must comply with the legal regulatory system according to Act N° 24.804 and its Regulatory Decree N° 1.390/98, meaning ARN's Regulatory Standards, regulatory requirements and sanctions, and with the international legal agreements according to what is established in the International Legislation accepted by the Argentine Republic on matters such as radiological and nuclear safety, security audits to nuclear material use, licensing and inspection of nuclear installations and international safeguards.

ARN based its opinion on the following issues:

1. CNA I is a second generation NPP whose design dates from the 60's and has been in operation since its commissioning in 1974 with a satisfactory safety level. The installation's operation for more than three decades, as well as the performance follow-up of other similar installations at the international level, allows the acquisition of great operative experience that helped, in turn, to implement significant safety back-fitting improvements.
2. CNA II has much more advanced safety aspects than its predecessor CNA I, coming from the "Konvoi" design concept that was used.
3. Apart from these original design safety aspects, the operative experience of CNA I and the applicable international experience should be added, such as improvement of the reactor's internal design and the elimination of "stellite-6" in the core materials.
4. CNA II is the first NPP where the licensing process is 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 carry out the regulatory tasks of evaluation, inspection and audits. The counselling is performed – as normal practice by ARN – through contracts or specific agreements. In fact contracts have been made with institutions in Argentina, USA, Germany and Canada (see article 3.8.).

3.18.3.2.1. CNA II components preservation

During the time that the construction of CNA II was postponed, one of the principal concerns was the appropriate components preservation. The following criteria were used:

- Influence on nuclear safety.
- Economic value.
- Replacement feasibility.
- Impact on project schedule.
- Preservation cost versus replacement cost.
- Damage sensitivity.

Component preservation tasks were divided in:

- Routine: Applicable to all SSCs. Their scopes and execution frequencies were defined in specific procedures and instructions.
- Non-routine: Applicable in function of the results of the routine preservation tasks.

In some occasions it was necessary to implement some corrective actions for conservation purposes, due to the results of routine tasks, external or internal assessment, or improvements in preservation criteria.

Important issues were taken into account for the materials, parts and elements affected by ageing. The most important were gaskets, rings, electrode weldings, greases and lubricants, glues and adhesives, NDT consumables, paints (civil and mechanical), fire proof mortar, gratings, supplies for the first filling (lubrication oils, control fluid oil), spray insulation for HP turbine casing, electrical and I&C components containing electrolytic capacitors.

The materials, parts and elements affected by new regulations were also taken into account for corrective actions, such as the insulation material for primary system components (experiences recently gained from sump clogging), change of refrigerant in chilled water machines and elements containing asbestos.

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 has 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. The tasks included visual inspections of reactor coolant, pressurizer, moderator and main steam pipes. Also, the main coolant pump and moderator heat exchangers were visually inspected. All the inspected components and pipes had satisfactory results showing their fitness for installation and future operation. Only minor findings were reported.

In the case of the bolts belonging to the pressurizer flange connection, a dye penetrant test was carried out with satisfactory results.

A destructive test in the electrodes for the welds in the reactor coolant line was carried out with the main purpose of verifying its weldability. The test results were satisfactory.

3.18.3.2.2. 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.

The current status concerning each critical issue is shown in the following subsections 3.18.3.2.2.1. to 3.18.3.2.2.4.

3.18.3.2.2.1. 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 y QSP 4a and 15a/c Rev. 2 mentioned in PSAR of CNA II).

In addition, in its updating process the following standards and guides were taken into account: IAEA Practice Code 50-C-Q, ISO Standard 9001:2000, Regulatory Standard AR 3.6.1., IAEA-GS-R-3, 2006, Occupational Health and Safety - OHSAS 18000 (IRAM 3800-Safety and Occupational Health) and ISO Standard 14000 Environmental Management System for Environmental Management.

In 2010 NA-SA sent to ARN a new version -QAP 115 Rev.4-. The application of this document is continuously audited by NA-SA and by the Regulatory Body in an independent way.

3.18.3.2.2.2. Design Authority

With the purpose of meeting this requirement and assuming the project direction, during the years 2004 and 2005 NA-SA maintained negotiations with the plant's original designer –Siemens- in order to discuss the technical and financial conditions to resume and finalize the construction of CNA II.

According to the negotiations, NA-SA UG-CNA II (Management Unit) has the responsibility for finalizing the basic and detail engineering, the construction and the commissioning of CNA II. Cooperation agreements with Siemens and AREVA were signed in order to fulfil this objective.

The CNA II detail design was completed by NA-SA with the technical assistance of different institutions. Some of them are shown in the following Table.

INSTITUTION	SUBJECT
SIEMENS	Provides supplies and services for the conventional area of the plant.
AREVA	Provides experts for the project in engineering, licensing, erection and commissioning areas.
AECL	Provides experts, engineering packages and supplies.
IAEA	A technical cooperation agreement was signed between CNEA-NA-SA and IAEA. The objective of the missions was an independent revision of the engineering and licensing activities. During 2006 and 2007 two missions took place, the first one regarding the Updating of Technological and Safety for CNA II and the second one regarding the analysis of state of preservation of stored components and demonstration of fitness for continued use. During 2008-2009 two expert missions were performed with the objective of analysing and making recommendations about the situation of the Design Authority.
UNIPI	Development of a platform for thermal-hydraulic design issues and deterministic safety technology. The scope for the platform is constituted by the thermal-hydraulic design and safety environment connected with CNA II construction and licensing. The word 'thermal-hydraulic' includes any issue that needs a thermal-hydraulic support, e.g. structural mechanics, neutron kinetics, radioactivity release and PSA. Signed end of 2006.
GRS	Assists to perform a PSA Level 2 for CNA II with the necessary support of NA-SA. The PSA Level 2 is together with PSA Level 1 and Level 3 (which will be both performed by NA-SA) a basic requirement for the licensing of CNA II to verify the Argentine Regulatory Standard AR 3.1.3. compliance. Signed 2007.
CEN/SCK	Integral solution for the surveillance program of the RPV. Signed 2007.
CNEA	A strategic partnership between NA-SA and CNEA was performed. CNEA specialists are delegated to NA-SA to perform activities in different areas such as engineering, licensing, erection and commissioning.
EMC2	Performs an evaluation of the break opening time for a 2A break LOCA.
TECNATOM	A contract for a Full-scope simulator for CNA II was signed at the beginning of 2010.

As was mentioned before, two IAEA expert missions visited Argentina (in 2008-2009). The objective was to analyse and make recommendations about the situation of the Design Authority according to the INSAG-19 "Maintaining the Design Integrity of Nuclear Installations throughout their Operating Life". For the recommendations fulfilment, NA-SA and the UG CNA II performed the organization of the Design Authority (DA).

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 (CRMED) 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. In addition, the Engineering Coordination of the Operation Unit is responsible for controlling the configuration of NPP's design through the organization and maintenance of drawings, technical specifications, manuals, design guides, engineering calculations, data bases, theoretical bases of CNA II's SSCs, in order to ensure that the knowledge base of design is preserved and expanded by the experience to be gained during commissioning and operation, also ensuring the identification of Responsible Designers and other external support.

Therefore, the DA's configuration adopted by NA-SA, in its role as CNA II's Licensee is carrying out a systematic treatment of its central objectives, which are the design changes' control and the design integrity maintenance, according to what it was above described.

To accelerate the design changes control management, a procedure was developed and approved as a management tool, regulating the functioning of the CRT for the treatment of issues related to CNA II, incorporating in its scope the aspects related to the approval of design modifications, evaluation of the mandatory documentation and the analysis of relevant events during the commissioning stage.

In this context, seven relevant design changes have been identified and documented. Three design changes have been approved by ARN:

- *I&C modernization.*
- *H2 recombination (PAR).*
- *Reactor controls update.*

Three are under evaluation:

- *Modification of the fast boron injection system (JDJ).*
- *Safety injection pumps strainer (JND).*
- *RPV surveillance program.*

The remaining design change is related with the LOCA 2A break concept and is pending of treatment and approval by the CRT.

According to what is stated above, concerning design changes' control, it can be concluded, that the configuration adopted for the DA as well as the methodology defined to regulate their management, are satisfactory to the Licensee and will allow complete the licensing process in the terms required by the Regulatory Body.

Regarding the design integrity maintenance, the transference from the Construction Group to the Operation Group is in process, being planned that the activities will be no affected, during Phases A and B of the CNA II commissioning.

Although the transfer of the related technical documentation will be carried out in a date closer to the plant nuclear commissioning start up, both groups are maintaining weekly meetings in order to agree the criteria and general terms of such transference in a systematic and ordered manner.

The Licensee has been considering the possibility to implement indicators of the DA performance. However, it was decided to postpone this initiative and reconsider it later during the operation stage to use the experience acquired during the construction and commissioning stages.

3.18.3.2.2.3. Review of the licensing basic criteria

In November 1977, the Regulatory Body signed with KWU enterprise (precursor of ENACE as Designer and Constructor of CNA II) 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.

Consequently for CNA II's design the LOCA evaluations were carried out in the frame of a probabilistic risk analysis, out of which arose, as design base accident, an instantaneous rupture in the biggest pipe connecting the primary circuit and the pressuriser (surge line), which is smaller than the break 2A.

From the Probabilistic Safety Analysis applied to a large LOCA (as the 2A break), a frequency less than 10^{-7} /year was obtained. That value was obtained by extrapolation, from the Biblis B NPP (German) taken as reference. The adoption of that value to the Argentine regulatory criterion (based on risk), made the designer consider that it was unnecessary to design the safety systems to cope with a 2A break of the main primary piping.

The Regulatory Body objected the bases of the risk analysis, as it was not possible to make any judgment on the validity of the extrapolation made, as the information supplied by the designer on the risk analysis was indicative rather than demonstrative. The Regulatory Body required more information on the LOCA 2A probabilistic evaluation.

Finally a probabilistic safety analysis including large LOCA was delivered to the Regulatory Body in April 2013 which has considered models' improvement and new sequences with different break locations.

Besides, it is worthwhile mentioning that the conclusion of a review of the present predominant basic criteria all over the world was 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). This was formalized in a regulatory requirement, by which, the utility has to demonstrate that the safety systems can cope with the consequences of such BDBA. Also the Regulatory Body has considered acceptable to change from an unrealistic 2A LOCA instantaneous scenario to a more realistic approach, where materials with high ductility and fracture toughness take some time to develop a rupture.*

3.18.3.2.2.4. Review of safety aspects

CNA II is a PHWR so its original design involves a positive void reactivity coefficient. For that reason ARN considered that it is of high importance to analyze 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.

A thorough analysis of this problem considers the following aspects:

- 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. Evaluation of the Break Preclusion (BP) Concept.
 - b2. Calculation performed based on adequate and validated codes.
 - b3. Improvements that may be performed to the design of the fast shutdown system.
 - b4. Improvements in the design in order to get a better void reactivity coefficient.

The analysis is being carried out by NA-SA under ARN request. Associated with these points the following may be pointed out:

a. Compliance with the regulatory criterion defined in Regulatory Standard AR 3.1.3.

- *The AR Standard 3.1.3. stipulates for the CNA II licensing that the intersection of frequencies and consequences of severe accidents sequences have to be below certain threshold.*
- *The methodology established in the above mentioned Regulatory Standard is based on the PSA L3, which uses the results of the PSAs L1 and L2. In the PSA L1 has been identified the accident sequences leading to core damage, being quantified their occurrence frequency by using the techniques of event trees and fault trees, with strong support in deterministic simulations of the evolution of the plant with the RELAP5 code.*

The PSA L2, takes the results of PSA L1 as input data, considers the severe accident progression through a probabilistic modelling (event trees containment accident progression) and deterministic (MELCOR code simulation of the severe accident sequences). The release's paths to the atmosphere are identified and the respective source terms as well as the annual occurrence probability are calculated.

The PSA L3 uses the results from the PSA L2, grouped into categories called release categories and, evaluates the atmospheric dispersion of the source terms associated with each release category and the dose to the public, through the code WinMACCS. Finally the maximum radiological risk to the public for each release category, which is necessary for the verification of the Regulatory Guide AR 3.1.3., is calculated.

- *A first complete version of the PSA L1 was delivered in September 2011 and the final version which included the ARN review was accomplished by the Licensee in October 2012. NA-SA has fulfilled the ARN's requirements for PSA L1 for the Commissioning License.*
- *The PSA L2 was also delivered in July 2011 and a new revision including the ARN comments was also accomplished by the Licensee in April 2013. Currently PSA L2 is being reviewed by ARN.*
- *The PSA L3 was delivered in August 2011 and the corresponding revision was also sent to the ARN in April 2013 and is being reviewed by ARN.*

b. Application of good international practices that arise from the following analysis

b1. Evaluation of the Break Preclusion Concept

It comprises principally:

- Evaluation of the design criteria based on the principles of leak before break.
- Revision and analysis of the applied concept of break size.
- Analysis of the time of opening of rupture 2A.
- Adequate SSCs mechanical design.
- Main components and piping, stress analysis, fracto-mechanics, NDE.
- Adequate Inspection Service Program.
- Detection of leakage.

The original Break Concept was enlarged including as a BDBA, breaks defined by a 2A's LOCA. This concept requires updating the application of the Break Preclusion (BP) in those lines as defined by NA-SA and for which a leak before break (LBB) behaviour is assumed.

The demonstration of acceptable BP follows the rules of KTA 3201.1/2/3 in order to fulfil the pre-requisites for the basic safety concept and the KTA 3201.4 as independent measures redundant.

The updating demonstration is being jointly performed by AREVA and NA-SA. When applying BP, issues such as flanges have required specific approach in order to demonstrate that flanges don't introduce singularities that could jeopardize the BP behaviour.

Both leak detection system as well as the ISI is parts of the independent redundancies. Regarding the leak detection system, specific considerations were required by the Regulatory Body in order to manage the different sensibilities required for the pipes of heavy water and light water. Also, has been required to show that the instrumentation have the appropriated environmental qualification coming from design basis accidents (DBA).

In relation to the ISI program and the corresponding pre-service, NA-SA redefined the basis in order to have a program consistent with the design. As a consequence, an updating of the ISI program is in progress. In this sense, for those pipes with BP concept, KTA 3201.4 was used and, for the rest were followed the KTA 3211 and the ASME Section XI.

Stress and Fatigue analysis for the main components are also in progress through different contracts with AREVA and foreign and local subcontractors. NA-SA has installed the fatigue monitoring system (FAMOS) in order to monitor thermal fatigue and to confirm that the plant continues under the licensing basis during operation.

The BP concept is used to manage the consequential failures, as jet forces. German practices in this regards consider designing the protective measures following the design basis, such as 0.1A break. NA-SA was working in a redefinition of the protective measures, especially in the reactor coolant line, for which the protective measures are dimensioned using the maximum detectable flaw. This approach was submitted to the Regulatory Body which accepted the above mentioned criterion due to the conservative value adopted.

The package of documentation that complete the BP demonstration will be finished before the Commissioning License will be granted. The Design Authority must review the updated BP concept as a pre-requisite for the regulatory review.

Regarding to the revision and analysis of the applied concept of break size, in order to determine a realistic break opening time, NA-SA contracted to EMC2 who has finished the analysis including the test for CNA II's material, a complete model of the systems primary and moderator of CNA II by using a program finite elements (FE) and the seismic evaluation.

Concerning the coolant leakage detection, an update of the tritium monitoring system is already installed.

b2. Calculation performed based on adequate and validated codes

In order to optimize the design of the power distribution control system scheme between a plant simulation code (DYNETZ), a spatial neutronic kinetics code (PCE) and a two-phase thermohydraulic code (RELAP) was developed.

The whole plant thermohydraulics and the protection, limitation and control system are simulated by DYNETZ, except for the reactor core that is simulated by the 451 nodes RELAP nodalizations, where each node representing an individual coolant channel from the thermohydraulic point of view.

The power distribution in the core is calculated by the 3D neutron kinetic code PCE in the 451x4x23 nodes representing the Fuel Elements. This computing tool is capable of predicting power distribution transients originated by possible sub-cooled nucleate boiling at the end of the coolant channels, by Xenon-135 transients, by control-rod malfunctions, etc.

As was mentioned in subsection 3.18.3.2.2.2., NA-SA has signed an agreement with UNIPi for the development and application of a model of CNA II utilizing Relap5/3D- Nestle (thermo-hydraulic and 3D-neutronic kinetic coupled codes) and CFX (Computer a Fluid-Dynamic program) to evaluate AOO, DBA and BDBA).

Additionally, an international "ad-hoc" expert group (IRG – International Review Group) had supervised and approved all the activities developed by UNIPi for NA-SA. Besides it established a connection between the scope of these activities and:

- Whether or not experimental tests that may arise as result of this analysis are necessary.
- The probabilistic safety analysis.
- The counselling to NA-SA due to eventual requirements or requests to extend the information for ARN.

The above mentioned accident analysis were already performed. The CFD results for the distribution of the boron injected by the Fast Boron Injection System in the moderator tank were validated using experiments performed by the original designer (Siemens-KWU) during the 90's. They consisted in a Plexiglas moderator tank including the coolant channels (scale 1:7.666). Different lance designs and the corresponding boron cloud concentration distribution inside the moderator tank were investigated.

During an instantaneous LOCA, a power excursion takes place which is shut down by the fast boron injection system (JDJ). The results of the previous computational fluid dynamic (CFD) analysis were used to determine the boron negative reactivity in the 2A break LOCA calculation. Finally, the 2A Break LOCA analysis were performed using the RELAP5/3D- NESTLE (TH and 3D-NK coupled codes).

The objective of this analysis was to determine the needed delay time for the Fast Boron Injection System to cope with this accident. The efficiency of the Safety Systems (Emergency Core Cooling and Reactor Shutdown) was assessed with these coupled codes. Detailed analysis is presented in Chapter 15 of the FSAR.

In addition, ARN has carried out independent assessments through an agreement with the Purdue University based on the development and application of a CNA II model using the Relap5 Mod3/Parcs, provided by US-NRC.

b3. Improvements that can be made to the design of the fast shutdown system

According to the updated licensing basis (as stated in last paragraph of section 3.18.3.2.2.3.), NA-SA was required to extent the performance of the safety systems with the main purpose to cope with the BDBA's consequences.

In order to fulfil this requirement, design changes in the Fast Boron Injection System with the objective to reduce the time injection were implemented. The target time was the result between system reliability and structural integrity compatibles with available engineering solutions.

A testing facility was constructed in order to verify these modifications. This facility (scale 1:1) used the original components of the system which were later placed in the plant and the layout was similar to the longest train in order to have a conservative result. The discharge of the system took place in a reservoir simulating the conditions of the moderator tank. The tests were performed during 2011 - 2012 period and the results were acceptable regarding the time injection optimization. The Licensee still must submit to ARN the results of the break opening time from the analyses described above in order to verify the sufficiency of the performed design modifications.

The design modifications were reviewed and approved by the Design Authority before they were sent to ARN. As part of the documentation reviewed, consideration regarding the modified system efficiency was included that the reactivity's insertion occurs before in time and with a higher magnitude. Therefore, the efficiency study performed by the designer is still valid for the modified system.

With the purpose to decrease even more the time injection, NA-SA has also modified the I&C related with the fast boron injection system using a modern technology. So far, this I&C was installed in the plant as a test mode with the purpose to get experience.

b4. Improvements in the design in order to get a better void reactivity coefficient

NA-SA is analysing an eventual modification of the original fuel element design in order to reduce the positive void reactivity coefficient (or make it completely negative). For the eventual modification of the fuel element NA-SA makes use of the experience acquired in CNA I during the project ULE (where the fuel element material was changed from natural uranium to slightly enriched uranium).

3.18.3.2.3. Current design and engineering tasks

Related to the original design, the following relevant activities were completed:

- Design verification of the mass flow distribution in the lower plenum.
- Redesign of the sump filtering systems.
- PARs were installed inside the containment to prevent containment failures due to hydrogen explosions during a severe accident.
- Improvement of the Fast Boron Injection System reducing the injection delays and increasing the injection velocities.
- *I&C and Reactor Control update.*

Related to the engineering tasks, the following relevant activities were completed:

- Piping detail engineering.
- Completion of missing supports and anchorage plates.
- Analysis of the consequential failures after a pipe break. *Dimensioning of the protective goals.*
- *Redefinition of the surveillance program.*
- *Seismic margin assessment including updated seismic hazard analysis for Atucha site.*

- Stress and fatigue analysis in pipes and components missing at the time of the project resumption.
- Detail engineering for the 500 kV sub-station.
- *Cabling engineering.*
- *Pre-service inspections of piping and components.*

3.18.3.2.4. Current construction tasks

The following table shows the Main Construction items – June 2013 Progress Status

<i>Civil</i>	<i>100%</i>
<i>Piping</i>	<i>100%</i>
<i>Mechanical Equipment</i>	<i>100%</i>
<i>HVAC</i>	<i>98%</i>
<i>Electrical Equipment</i>	<i>98%</i>
<i>I&C</i>	<i>55%</i>

At the present time the erection and construction activities are almost finished. The commissioning tests of the SSCs are under progression and will be finished by the end of 2013. The fuel elements are already loaded in the reactor core and some hydraulic commissioning tests will be performed by the third 2013 quarter. After the finalization of these activities, the light water has to be replaced from the primary system and the moderator system by refilling it with heavy water. At this stage the plant will be ready to initiate the nuclear commissioning tests.

In the areas of electrical and I&C, the remaining construction activities are cabling, support installation, process and radiation instrumentation.

The transference activities imply the transference of 556 packages from the construction to commissioning, as defined by each System Engineer Responsible. Each of these packages includes mechanical components, electrical (process) and I&C (logical). At June 2013 a total of 80% of packages were already transferred.

Regulatory activities carried out at CNA II include evaluations and inspections aimed at verifying the licensing basis. In the case of inspections they are part of a proactive licensing process and aim to demonstrate that all measures taken during construction and assembly have been conducted in compliance with the requirements outlined in the construction license and specific requirements, the applicable codes, standards, and specifications. The regulatory inspections cover the areas of Mechanics - Materials, I&C, Electrical and Processes, and are focused on plant systems relevant to safety (safety systems and safety related systems) according to their importance for nuclear and radiological safety.

The assessments, also in the areas of Mechanics - Materials, I&C, Electrical and Processes, were targeted primarily to verify the design basis and evaluate deviations from the state-of-the-art on the subject, with the intention to make recommendations to the Licensee in case to be necessary for the safety objectives achievement.

In addition, both the assessments and inspections took into account the operating experience gained in CNA I and other NPPs.

ARN is also pursuing inspections and evaluations as part of regulatory activities to control the on-going preliminary tests.

3.18.3.2.5. Design improvements implementation

As were mentioned in sections 3.14.3.1.1.; 3.14.3.1.2.2.; 3.16.2.9. and 3.17.2.3.3.1.; 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 measures / changes have been foreseen to be implemented to prevent 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.*

- *Connection among the DGs of CNA I and CNA II.*
- *Dose rate remote measurement system.*
- *Alternative power sources (MDGs).*
- *Extension of the batteries availability.*
- *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).*
- *Cooling of the RPV external side.*
- *Venting filtered containment system.*
- *I & C improvements to provide information for severe accident management (beyond design basis accidents).*
- *Alternative cooling mode of the DGs.*
- *Disconnection of electrical loads.*
- *Increasing DGs autonomy.*
- *Switchgear building ventilation.*

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. requirements related to man-machine interface are in general fulfilled.

The Dry Storage of Irradiated Fuel Elements System is composed *by a transference cell; transference flask and the silos field*. The capacity of each silo consists of 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 216. *The stored inventory at the end of 2012 was 99,900 fuel elements in 185 silos.*

3.18.3.3.1. Design improvements implementation

As were mentioned in sections 3.14.3.1.1.; 3.14.3.1.2.3.; 3.16.2.9. and 3.17.2.3.3.1.; 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 measures / changes have been foreseen to be implemented to prevent beyond design basis accidents or to mitigate their radiological consequences:

- *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.*
- *Relocation of the site where is located the centre of the plant lighting system.*

3.18.3.4. CAREM REACTOR PROTOTYPE

The concept of CAREM that belongs to the very low or low power nuclear plants was put forward from the very beginning as an advanced designed reactor, being the precursor of innovative concepts as regards safety. *CAREM* is a light water reactor with new design solutions, which contributes to its high level of safety, being the followings its main innovative aspects:

- Integrated Primary System.
- Self-pressurization.
- Passive Safety Systems.

Even though both technical and engineering solutions associated to the NPP's technology and, the innovative design characteristics are correctly verified during the design phase, it was considered convenient to construct a reactor prototype to validate its design, manufacturing, installation and operational aspects as well as verification SSC's reliability. A more detailed description may be found in Annex V.

Due to this fact, CNEA, as the primary agency to conduct nuclear science and technology developments in Argentine Republic, proposed, to the National Government, to carry out the construction of the CAREM Reactor prototype, 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 30 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. Construction of CAREM prototype

By means of the Presidential Decree N° 1.107/2006, CNEA have been instructed to follow the necessary actions for the CAREM's prototype construction and to assume the responsibility for the construction, commissioning and its further operation.

CNEA, as the owner of the technology *and Design Authority* of the CAREM Plant, *is responsible for* looking after safety matters as well as planning and construction solutions. CNEA provides scientific and technological backup to the works related to the project and is responsible for the proper implementation of tests and qualification of fuel elements and safety components.

In line with the integration of national capacities, the participation of domestic companies is preferred, such as INVAP S.E., Industrias Metalúrgicas Pescarmona S.A. - IMPSA, NA-SA, among others, which have the technological capacity to cover engineering and manufacturing of heavy components and contribute to the project implementation, are being selected.

CNEA has fostered three companies with great technical capacity in the nuclear area that will play an important role in the CAREM-25 Project. These companies are CONUAR, FAE, DIOXITEK and ENSI.

It is expected that its construction will be finalized by the end of 2016, 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 provisions were made until the finalization of the project.

During the 2007-2010 period, the following tasks were conducted: 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.

A test loop for the drive mechanisms and other safety components, performed with IAEA support, *has been operated since August 2012.*

The Project organizational structure was completed and the personnel dedicated to this project are composed by 170 technical persons with full time dedication and 70 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. Licenses and siting

The Chamber of Deputies of the Buenos Aires Province declared its interest to have the CAREM-25 prototype constructed in Lima, Province of Buenos Aires, in the lands of CNEA, next to Atucha I and Atucha II NPPs.

Taking into account that CAREM reactor was considered a prototype of innovative design; the ARN has established an ad hoc licensing scheme, applicable to the construction and commissioning stages. In this sense, the Regulatory Body has established a licensing scheme applicable to prototypes for new nuclear reactor designs. This scheme provides for licensing in successive stages, the first of which corresponds to obtaining the Authorization for Use of Site and Construction. The requirements for obtaining such authorization consist of the presentation of information on the following aspects:

- *Environmental Radiological Impact.*
- *Waste Management.*
- *Radiological Emergency.*
- *Design Information.*
- *Quality management.*
- *Project Schedule.*

The CNEA submitted the information requested by the Regulatory Body and is currently in the final review stage. It is estimated that the Authorization for Use of Site and Construction will be issued in the course of this year.

As was mentioned in section 3.17.2.3.2. of this National Report, the site selected is a part of the Atucha site where two NPPs are already located (*CNA I and CNA II*) which includes the necessary infrastructure and effective security measures with an established response force, that has proven its competence to handle emergencies effectively during annual drills.

The site is suitable to build another NPP as was demonstrated in several studies carried out during the CNA I Project. These studies include, inter-alia, external events, population density & distribution, and NPP location site lay out and are complemented by others related to the CAREM specific location, such as the geological studies and the mutual impact of CAREM on CNA I / CNA II, and vice versa.

Agreements are already signed to ensure co-operation and feedback of experience between CNEA and NA-SA in connection with radiological and nuclear safety, physical protection, security, and exchange of technical information including environmental monitoring data.

3.18.3.4.4. Critical Technical Issues Established by ARN

Due to the fact that CAREM 25 is the first prototype of a small and medium reactor (SMR) of innovative design, ARN has adopted a progressive approach to the CAREM's licensing process (ad hoc licensing scheme) to be applied to the stages of design, construction, commissioning and testing, with specific characteristics, aimed to achieving the required safety performance. ARN considered the following critical technical issues:

3.18.3.4.4.1. Design Authority

CNEA has fostered the CAREM Project as the organizational structure responsible for the design and construction of the plant. CAREM Project assumes responsibility as DA.

3.18.3.4.4.2. Review of the Licensing Basic Criteria

Within the licensing framework described in 3.18.3.4.4., ARN has required specific documents in addition to the Preliminary Safety Report, which includes the earlier information regarding the development phases of the design, validation and testing during all the Project stages to assure a close involvement of ARN in those issues relevant to regulatory purposes.

3.18.3.4.4.3. Review of Safety Aspects

Both design concept as well as the safety features evaluation is in process. The Preliminary Safety Report's evolution is in a mature stage after a 5-year-review process. Last trends in nuclear safety and safety criteria have been incorporated in CAREM design during this process.

3.18.3.4.4.4. Design and Engineering Tasks

The civil detail engineering of the nuclear module is already complete. An independent structural verification is being prepared.

The engineering of the safety and auxiliary systems, the reactor protection system and the control system are on-going.

A bid for the RPV's procurement is under process.

3.18.3.4.4.5. Construction Tasks

There are site preparation activities on-going. The excavation of the nuclear module has been completed. A contract for the nuclear module civil construction has been signed.

3.18.3.5. FOURTH NUCLEAR POWER PLANT

Currently different technological alternatives are under study for the construction of the Fourth NPP in Argentina. The installation, of a module of approximately 1000 MWe is estimated. The location has not yet been determined; although the Atucha site has good chances of being selected.

3.18.4. COMPLIANCE WITH THE OBLIGATIONS IMPOSED BY THE CONVENTION

In Argentina, NPPs have been designed and constructed in such a way as to have several reliable protection levels, in order to prevent the release of radioactive materials to the environment, prevent accidents and mitigate their consequences in case they occur.

Therefore, the country complies with the obligations imposed in Article 18 of the Convention on Nuclear Safety.

3.19. ARTICLE 19: OPERATION

Each Contracting Party shall take the appropriate steps to ensure that:

- i. The initial authorization to operate a nuclear installation is based upon an appropriate safety analysis and a commissioning programme demonstrating that the installation, as constructed, is consistent with design and safety requirements;*
- ii. Operational limits and conditions derived from the safety analysis, tests and operating experience are defined and revised as necessary for identifying safe boundaries for operation;*
- iii. Operation, maintenance, inspection and testing of a nuclear installation are conducted in accordance with approved procedures;*
- iv. Procedures are established for responding to anticipated operational occurrences and to accidents;*
- v. Necessary engineering and technical support in all safety-related fields is available throughout the lifetime of a nuclear installation;*
- vi. Incidents significant to safety are reported in a timely manner by the holder of the relevant licence to the ARN;*
- vii. Programmes to collect and analyse operating experience are established, the results obtained and the conclusions drawn are acted upon and that existing mechanisms are used to share important experience with international bodies and with other operating organizations and regulatory bodies;*
- viii. The generation of radioactive waste resulting from the operation of a nuclear installation is kept to the minimum practicable for the process concerned, both in activity and in volume, and any necessary treatment and storage of spent fuel and waste directly related to the operation and on the same site as that of the nuclear installation take into consideration conditioning and disposal.*

3.19.1. INTRODUCTION

The ARN authorized NPPs commercial operation on the basis of the judgements mainly supported by both design safety assessments and commissioning follow-up results at the installations.

Once in operation, NPPs are operated by the Licensee according to what is established in the Operating License, as well as Mandatory Documentation i.e. Operational Limits and Conditions set in the SAR and the Policies and Principles Manual. This last document is based on the technical specifications of the installation and the operating experience.

ARN inspectors verify that the Mandatory Documentation that includes the three above mentioned documents are fulfilled. Furthermore, as part of routine inspections, resident inspectors audit and control procedure fulfilment, regular test performance, ISI, programmed maintenance and any other safety related activity.

The Reactor Manager is supported by an engineering section providing part of the technical support needed for the NPP operation. In addition, the Licensee also has an engineering division satisfying some of the technical support the installation needs. In order to cover other required services, domestic or international contractors are used.

The process, through which the installation operating experience feedback is carried out, both at the Reactor Manager and at the Licensee level, must comply with the Operating License requirements and Regulatory Standard AR 3.9.2. as well as with other applicable regulatory requirements.

The feedback process of operating experience of domestic NPPs involves the following entities: Licensee, ARN, Designers, Component Suppliers and international organizations dedicated to information distribution.

Furthermore, NPPs have programs for fire protection and management of radioactive wastes generated during their operation. The later program includes low and medium radioactive waste treatment and its subsequent storage.

3.19.2. INITIAL AUTHORIZATION TO OPERATE

3.19.2.1. CNA I - INITIAL AUTHORIZATION TO OPERATE

In May 31, 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 - INITIAL AUTHORIZATION TO OPERATE

CNE's initial authorization was issued according to the requirements established in Regulatory Standards AR 3.8.1. and AR 3.8.2. The first one is related to pre-nuclear commissioning and establishes that the Licensee must have a Pre-Commissioning Program and an Organization to carry it out. The pre-nuclear commissioning program comprises those tests required to demonstrate the safe operation of the NPP.

Regulatory Standard AR 3.8.2. also establishes that the Licensee must have a nuclear commissioning program and an organization to carry it out. The standard also establishes that the Licensee must 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 has the main responsibility for evaluating each of the stages the commissioning program is divided into, and authorizes the transition from one stage to the other.

During pre-nuclear and nuclear commissioning stages, the ARN verified that the Licensee complied with the mentioned standards.

3.19.3. 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 - OPERATIONAL LIMITS AND CONDITIONS, MAINTENANCE, AND TESTING

The conditions for the authorization of the commercial operation of CNA I were established in the Operating License. The main requirements for the NPP, 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.

Initially in CNA I, there were no specific document referring to operational limits and conditions, as there are in most of the NPPs. The existing information (at that time distributed in different documents, such as the SAR, the Operating Manual, and the Maintenance Manual) have been collected in the Policies and Principles Manual that establishes the ranges of valid values some plant operational parameters must comply with, other specifications, 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 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 program including ISI activities related to significant components, equipment and systems are routinely carried out, mainly involving the reactor pressure vessel (RPV); primary, moderator and volume regulation systems, as well as steam generators (SGs) and moderator heat exchanger tubes.

As was informed in the Fourth Nuclear Safety Convention (NSC) Report, CNA I's Operational Limits and Conditions were modified regarding upper allowed values for some parameters. These changes were included into the corresponding operating procedures.

Additionally, a CNA I ISI Manual upgrade was performed and an important review of the Periodic Test Procedures was carried out. Some more specific specifications were added, mainly related to test acceptance criteria.

3.19.3.2. CNE - OPERATIONAL LIMITS AND CONDITIONS, MAINTENANCE, AND TESTING

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).

Since its commissioning, CNE has a Policies and Principles Manual where operational limits and conditions for the safe operation of the installation are established. 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 will not be framed in the Manual of Policies & Principles currently in force. Therefore, the licensee was authorized by ARN to modify this Manual to incorporate the corresponding guidelines to govern the operating conditions during this period.

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 program including ISI activities related to significant components, equipment and systems are routinely carried out, mainly involving the following: pressure tubes; primary, 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

The criteria and scopes of CNA I's and CNE's Inspection Programmes were described in detail in the previous NSC Reports. In the following paragraphs, the most important inspection activities performed during the last period in both NPPs are detailed.

3.19.4.1. CNA I - INSPECTIONS

3.19.4.1.1. In Service Inspections

The ISI in CNA I were performed during the reported period in agreement to a previously adopted schedule.

The inspections results were satisfactory as in previous inspection periods. In this way, the plant fulfils the inspection requirement.

3.19.4.1.2. RPV Internal Components Surveillance Program

Once CNA I's back-fitting was completed, there was a regulatory requirement for the development of a RPV internal components surveillance program, which started in the planned 2006 outage.

According to this program, twelve positions were selected to take out the corresponding channels for their inspection. In each programmed outage, four channels are taken out in order to inspect:

- In core neutron flux sensor guide tubes.
- Cooling channel foils.
- Moderator tank wall and bottom.
- The in core neutron flux sensor guide tubes that were plugged.
- The growth of the channel foils.

To these selected elements some others may be added as a result of the "early failure detection program".

As was mentioned in the previous National Report, during the 2006 scheduled outage, one cooling channel was withdrawn in order to extract samples to perform mechanical tests in hot cells. The corresponding final report carried out by CNEA was issued in 2010.

The surveillance program continued in the scheduled outages carried out in 2011, 2012 and 2013. From 2011 the inspection of upper nozzle welding (RPV top side) was included following WANO recommendations. In 2012 following the results of a workshop performed with AREVA, an additional survey consisting in the visual examination including nozzle's axial space in the neighbour channels of those intervened was carried out. AREVA has performed an independent assessment of the surveillance program results including the samples analysed by CNEA.

The conclusion is that RPV's internal components behaviour satisfies the designer's condition and, no further actions are necessary.

3.19.4.1.3. Fire protection

Regulatory Standard AR 3.2.3. establishes the safety criteria against fire (or events generated by it) and explosions resulting from fire that may affect a NPP's radiological or nuclear safety. These criteria include the stages of design, commissioning and operation of the installations. The fulfilment of the criteria is verified through inspections carried out by ARN inspectors and analysts.

A Fire fighting system inspection was carried out by the Regulatory Body with the participation of the radiological safety team of the Federal Agency's fire-fighting body. The most important inspection recommendations and the corresponding implementation were detailed in the previous CNS reports.

The personnel belonging to the fire brigade routinely performs an annual retraining in specialized fire fighting centres. The Regulatory Body controls and inspects the fire brigade retraining.

3.19.4.2. CNE - INSPECTIONS

3.19.4.2.1. In Service Inspections

The ISIs in CNE was performed as scheduled during this last period. Looking for a more orderly report of results, the most remarkable data obtained from inspections concerning pressure tubes, SGs and feeders are detailed in article 3.14.

3.19.5. OPERATIONAL PROCEDURES IN NORMAL AND ACCIDENTAL CONDITIONS

Most of the CNA I's operational procedures, either in normal or accidental conditions, are included in the Operating Manual. This document 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.

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 subsection 3.14.3.1.2.

3.19.6. ENGINEERING AND TECHNICAL SUPPORT

CNA I 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, and AECL, responsible for the design and construction of CNE, as well as the companies that operate CANDU type reactors, with which there is an active experience exchange.

During the reported period the engineering branch of the CNA I was reorganized including a Systems Engineering Division.

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 causes and effective remedial and corrective actions commensurate with the situations.

Furthermore, the Operating License sets particular conditions referred to the subject and some specific requirements have been issued concerning it.

It should be noted that since June 2011 the OPEX Program in CNA II has been implemented. This plant is currently under commissioning.

During the 2010-2012 period, 30 relevant events were reported by CNA I and CNE. A total of 25 root causes / contributing causes were identified of which six were related to human factors and 19 were related to equipment failure.

The most significant operational events in CNA I, CNA II and CNE and how the Licensee and the ARN acted are given in [Annex VI](#).

3.19.8. OPERATING EXPERIENCE

In order to improve operational safety in CNA I and CNE, a periodic analysis of their operating experience, and, to a smaller extent, an assessment of other NPPs operating experience are carried out.

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 other plant, to the ARN and to the international nuclear community through the IRS.

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 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.

As a result of the accident occurred at Fukushima Daiichi, the corresponding lessons learned were taken into account by both the Licensee and ARN. In this sense a stress test has been required by ARN to the Argentinean NPPs Licensee (including the NPPs in operation and under construction) in order to detect possible weaknesses and implement the corresponding improvements (see sections 3.14.3.1.1.; 3.14.3.1.2.; 3.16.2.9. and 3.17.2.3.3.).

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.

In order to gain experience from the CNA II commissioning stage, procedures for events report and analysis have been prepared and implemented. At present, the relevant failures or incidents that occur during the execution of preliminary tests (no nuclear) are analysed and the corresponding reports are submitted to the ARN and to the Commissioning Committee.

Therefore, the OPEX in CNA II is being used both by the Plant Operations group, as by engineers responsible for commissioning.

3.19.8.1. FEEDBACK FROM LOCAL OPERATING EXPERIENCE

CNA I 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 both NPPs the following internal events are detected, recorded and analysed:

- Significant events, defined according to the criteria set in Regulatory Standard AR 3.9.2.
- Unforeseen outages.
- Minor events (*low level events*) or reportable events in CNA I 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 I, any person belonging to the installation 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, proposes corrective actions and follows their implementation and diffusion. All the operational incidents, significant and minor events, their corrective actions and their follow-up are recorded.

In CNA I 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.

In CNA II, the CNA I OPEX program was adopted as a model, with the purpose of unifying the organization culture. In addition, reports and reporting procedures have been adapted and agreed so as to standardize the criteria for treatment and analysis in CNA I and CNA II. Currently, all the information generated internally in each plant is shared.

The significant events are communicated to the ARN according to Regulatory Standard AR 3.9.1. and afterwards an analytical report is issued according to a format established in the plant's Operating License.

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.*

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 VII](#).

Finally, and as an example of the OPEX feedback of the NPPs in operation in Argentina, the transfer to CNA II of the lessons learned from CNA I's operation should be mentioned.

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.

CNE has had, since the beginning of its operation, a fluent communication with other CANDU plants of similar design, such as Point Lepreau, Gentilly-2, Wolsong-I, in order to exchange operating experience. Moreover, it is member of the CANDU Owners Group since its creation.

Presently, both CNA I 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 heterogeneous and not always profitable, as it essentially depends on the characteristics of the plant's design.

CNA I has been using the WANO database since 1996. The collection, selection and classification of information have been systematised.

In CNA II, events from WANO (especially SOER) are referred to the Commissioning staff for information and consultation. Once the plant is delivered to the operator, analysts will be assigned to those that need it, so that CNA I and II work with the same standards regarding external events.

CNE uses COG databases as part of its usual working activities. 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.

ARN examines the effectiveness of operating experience feedback using information coming from national and international databases.

This information is analysed by an analysts 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.
- Application of root cause methodologies.
- Corrective actions.
- Corrective action follow-up.

As a consequence of a requirement issued by the ARN in 1998, the Licensee started a formal and systematic process of evaluation of the operating experience in order to obtain feedback to improve reliability and availability of the NPPs.

The Licensee prepared an "Operating Experience Management Program" in order to analyse events (at National and International level) to be used as a feedback of Operating Experience from domestic NPPs. The major actions required by the ARN include:

- Use of international and national databases.
- Use of root cause analysis methodologies in the cases where an event is applicable in domestic plants.
- Taking immediate corrective actions to avoid event occurrence or recurrence.
- Corrective action follow-up.
- Lessons learned from analysis.

The Licensee constituted three working groups: two within each plant site and the third within the Licensee headquarters, to obtain feedback to improve plant systems (modifications) and optimize maintenance activities (through the execution and follow-up of corrective actions).

The program prepared by the Licensee and presented to the ARN included goals to be reached, implementation procedures and the professional profiles of the working group staff. Emphasis was given to the improvement in safety of NPPs obtained from the feedback of operating experience.

To fulfil the program, a set of activities to be carried out was defined and trend analysis, workshops to share experience and training were also included. The Licensee prepares a quarterly report including

the results obtained from the application of the program. Besides, NPP's senior teams evaluate "low level events" and "near misses" creating their own database.

The Licensee working group performs a screening analysis using international databases selecting the applicable events for the domestic plants. After screening, those events are analysed in detail and they are presented to the plants for evaluation. The evaluation could include recommendations, proposed design modifications, changes in procedures and training courses for operators if necessary.

The *headquarters* program coordinator *group (OPEX group)* reviews more than *1,000* reports per year from different international Sources. However, due to design, procedures, systems and components or low safety significant actions, approximately only 10% of the events are applicable.

The Regulatory Body has performed audits to the CNA I's and CNE's Operating Experience sectors that showed improvements on corrective actions, implementation, organization of training meetings and discussions as well as an increasingly experienced operating personnel.

There are many improvement actions resulting from the feedback of the National Operating experience and Operating experience from Foreign NPPs in CNA I and CNE. Many of them are being used in the backfitting program of the plants. Examples were shown in the previous National Reports.

Regarding CNA II, the Commissioning Documentation Department wrote the operating instruction IO-19 "Operating Experience Program during Phase A and B of the commissioning of CNA II ", in order to implement this program from the commissioning stage. Since its entry into force in 2011, this instruction has defined the organization and an appropriate methodology to the phases A and B of the commissioning process to report, analyze and document systematically the treatment of internal and external events linked to the Operating Experience Program (OEP) which was already implemented.

The OEP is aimed at improving the plant operation's safety culture as well as the reliability of systems and equipment, after they have passed the commissioning tests.

The efficient use of the OPEX acquired in an organization implies a systematic analysis of the events that have been well documented in order to implement measures to prevent that these or others with similar cause happen again.

It is known and accepted that low-level events and those events with or without consequences, have the same causes that events with significant consequences, for which their proper reporting and analysis contributes to the prevention of significant events. The report of these events contributes to the continuous improvement of processes and staff training.

With the OEP implementation, it is expected that both the commissioning staff, as well as the operation and maintenance staff, promotes an open and critical attitude of the staff, so that each event considered relevant is properly reported and analysed, that the analysis are disseminated, that corrective actions are implemented and that the lessons learned are being utilized.

CNA II has a Data Base of OPEX that can be accessed by the Central OPEX Group of NA-SA and contains the following information:

- Applicable procedures.*
- Applicable forms.*
- List of internal event reports.*
- List of external event reports.*
- Final report of events.*
- Corrective actions monitoring.*
- Periodic reports.*

In CNA II, selected information is received by the Central OPEX Group from the WANO database, since 2011. The events that are on these databases have been classified, recorded and disseminated in the areas potentially involved in order to take advantage of lessons learned from them.

OPEX is one of the sources for design changes included on CNE planned refurbishment for life extension. During the period 2010 – 2013 it could be mentioned the following:

- The OPEX group (Licensee's headquarters) has selected 140 reports from international databases for analysis or consideration.*
- The plants' OPEX group received 146 reports. Approximately 12% of them have contained recommendations and the rest were sent for information as generic events.*

- *The plants' OPEX group sent 82 responses to the events reported, including 27 proposals of corrective actions.*

Examples of lessons learned from local and international operating experience in the 2010-2013 period are shown in *Annex VII*.

3.19.8.3. PEER REVIEWS AND ACTIVITIES BETWEEN THE LICENSEE AND WANO FROM 2007 TO 2010

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).

Both NPPs under operation, received WANO Peer Review (PR) Missions and the corresponding Follow-Up (FU) (CNA I NPP: PR in 2006 *and* 2012, FU in 2009; CNE: PR in 2007, FU in 2009). NA-SA participated in WANO programs through the WANO – PC (Paris Centre): Peer Reviews, Technical Support Missions, Assist Visits, Operating Experience Programs, Workshops and Seminars. Argentina has also provided specialists from both NPPs to participate in every WANO Program.

Main activities during the reported period are listed in Table 19.1.

Table 19.1.

PERIOD	HOST	PARTICIPANTS	ACTIVITY
<i>January 18-22, 2010</i>	<i>Wylfa NPP, UK</i>	<i>Embalse NPP Nuclear Safety Manager</i>	<i>Safety Culture TSM</i>
<i>Jan 30 - Feb 3, 2010</i>	<i>New Delhi, India</i>	<i>NA-SA's Governor / General Manager and President</i>	<i>WANO BGM</i>
<i>January 31, 2010</i>	<i>New Delhi, India</i>	<i>NA-SA's Governor / General Manager</i>	<i>45th WANO Paris Centre Governing Board</i>
<i>February 8-12, 2010</i>	<i>Tihange NPP, Belgium</i>	<i>Embalse NPP Technical Support to Operation</i>	<i>Human Performance TSM</i>
<i>March 1-5, 2010</i>	<i>Cattenom NPP, France</i>	<i>Embalse NPP Production Manager</i>	<i>Unit in Operation Performance TSM</i>
<i>March 11-12, 2010</i>	<i>WANO PC</i>	<i>NA-SA's WIO</i>	<i>42nd WIOs' Meeting</i>
<i>April 6-9, 2010</i>	<i>Prague, Czech Republic</i>	<i>Atucha I NPP Chief Supervisor</i>	<i>6th Workshop on Middle Management Leadership</i>
<i>April 10-16, 2010</i>	<i>WANO PC</i>	<i>Atucha I NPP Production Manager</i>	<i>Workshop on Work Management</i>
<i>June 14-19, 2010</i>	<i>Paluel NPP, France</i>	<i>General Mgr Assistant</i>	<i>Human Performance TSM</i>
<i>June 19-27, 2010</i>	<i>Nogent NPP, France</i>	<i>Atucha I NPP Preventive Maintenance Manager</i>	<i>Housekeeping Corrective Action Program TSM</i>
<i>June 25 - July 4, 2010</i>	<i>Vandellos II NPP, Spain</i>	<i>Atucha I NPP Safety Manager</i>	<i>2nd Workshop on Exchange of Experience on Radiological Protection</i>
<i>July 26-30, 2010</i>	<i>OLDBURY NPP, England</i>	<i>Atucha I NPP Aging Manager</i>	<i>Plant Aging TSM</i>
<i>August 9-13, 2010</i>	<i>Atucha I NPP, Argentina</i>		<i>Technical Support Mission on Root Cause Analysis, Trend Analysis and Risk Assessment</i>
<i>August 23-27, 2010</i>	<i>Embalse NPP, Argentina</i>		<i>Technical Support Mission on Human performance and coaching</i>
<i>August 30 - Sept. 17, 2010</i>	<i>Emsland NPP, Germany</i>	<i>Atucha I NPP I&C Manager</i>	<i>Peer Reviewer (MA)</i>
<i>September 9-23, 2010</i>	<i>QINSHAN 2 Unit 3, China</i>	<i>Atucha I NPP Assistant to Plant Manager</i>	<i>Observer in a Pre Start-up Peer Review</i>

PERIOD	HOST	PARTICIPANTS	ACTIVITY
<i>September 13-17, 2010</i>	<i>Atucha I NPP, Argentina</i>		<i>Technical Support Mission on Design modification</i>
<i>September 13-17, 2010</i>	<i>Embalse NPP, Argentina</i>		<i>Technical Support Mission on Engineering support - Organisational effectiveness</i>
<i>September 22-24, 2010</i>	<i>WANO PC</i>	<i>NA-SA's WIO</i>	<i>43rd WIOs' Meeting</i>
<i>September 26 - Oct. 4, 2010</i>	<i>Blayais NPP, France</i>	<i>Embalse NPP Technical Support to Operation</i>	<i>Human Performance TSM</i>
<i>October 4-22, 2010</i>	<i>Vandellos NPP, Spain</i>	<i>Embalse NPP Rad and Safety Manager</i>	<i>Peer Reviewer (RP)</i>
<i>October 4-22, 2010</i>	<i>Vandellos NPP, Spain</i>	<i>Embalse NPP Operation Manager</i>	<i>Peer Reviewer (TQ)</i>
<i>November 16-19, 2010</i>	<i>Olkiluoto NPP, Finland</i>	<i>Atucha I NPP PSA Head</i>	<i>Probabilistic Safety Assessment Workshop</i>
<i>January 17-21, 2011</i>	<i>Zhuhai, China</i>	<i>Atucha I NPP RP Engineer</i>	<i>Leadership Workshop for Middle Managers</i>
<i>March 24-25, 2011</i>	<i>WANO PC</i>	<i>NA-SA's WIO</i>	<i>44th WIOs' Meeting</i>
<i>March 28, 2011</i>	<i>WANO PC</i>	<i>NA-SA's WIO</i>	<i>Fukushima Accident Meeting</i>
<i>April 11-15, 2011</i>	<i>Embalse NPP, Argentina</i>		<i>Technical Support Mission on Radiation Protection</i>
<i>April 28, 2011</i>	<i>WANO PC</i>	<i>NA-SA's Governor / General Manager</i>	<i>49th WANO Paris Centre Governing Board</i>
<i>May 9-27, 2011</i>	<i>Sellafield, England</i>	<i>Atucha I NPP PSA Head</i>	<i>Peer Reviewer (ES)</i>
<i>May 23-26, 2011</i>	<i>Würzburg, Germany</i>	<i>HQ Communications Manager</i>	<i>7th Communication Expert Group Meeting</i>
<i>June 13-17, 2011</i>	<i>Embalse NPP, Argentina</i>		<i>Technical Support Mission on Systematic Approach to Training</i>
<i>July 18-22, 2011</i>	<i>NINGDE NPP, China</i>	<i>Atucha II NPP Commission Advisor</i>	<i>Operation Management TSM</i>
<i>August 29-30, 2011</i>	<i>Atucha I NPP, Argentina</i>		<i>Technical Support Mission on Pre Start-up Peer Review Process</i>
<i>August 29-31, 2011</i>	<i>Toronto, Canada</i>	<i>Embalse NPP Radiological Protection Manager</i>	<i>Radiological Protection WANO-COG Workshop</i>
<i>August 29-31, 2011</i>	<i>Toronto, Canada</i>	<i>Embalse NPP Radiological Protection Deputy Manager</i>	<i>Radiological Protection WANO-COG Workshop</i>
<i>September 5-9, 2011</i>	<i>Atucha I NPP, Argentina</i>		<i>Technical Support Mission on Work Management</i>
<i>September 12-16, 2011</i>	<i>Almaraz NPP, Spain</i>	<i>Atucha I NPP Safety Manager</i>	<i>Fire Protection TSM</i>
<i>September 22-23, 2011</i>	<i>WANO PC</i>	<i>NA-SA's WIO</i>	<i>45th WIOs' Meeting</i>
<i>October 17-21, 2011</i>	<i>Vandellos II NPP, Spain</i>	<i>Embalse NPP Technical Support to Operation</i>	<i>Human Performance TSM</i>
<i>October 24-28, 2011</i>	<i>Bugey NPP, France</i>	<i>Atucha I NPP I&C Manager</i>	<i>Coaching Manager in the field TSM</i>
<i>November 7-11, 2011</i>	<i>Angra III NPP, Brazil</i>	<i>Atucha II NPP Construction Manager</i>	<i>Target Organisation TSM</i>

PERIOD	HOST	PARTICIPANTS	ACTIVITY
November 14-18, 2011	Thiange NPP, Belgium	Atucha I NPP Production Manager	Work Management and Planning TSM
November 14 - Dec. 2, 2011	Saint Laurent NPP, France	Atucha I NPP Corrective Maintenance Manager	Peer Reviewer (MA)
November 14 - Dec. 2, 2011	Saint Laurent NPP, France	Atucha I NPP Assistant to Plant Manager	Peer Reviewer (OA)
November 21-25, 2011	Saint Alban NPP, France	Atucha I NPP Production Manager	Work Management and Planning TSM
December 12-16, 2011	Ning De NPP, China	Atucha I NPP Plant Manager	Exit Representative in the Pre Start-up Peer Review
February 29 - March 3, 2012	WANO PC	NA-SA's WIO	46 th WIOs' Meeting
April 2-20, 2012	Chinon NPP, France	Atucha I NPP Radiological Protection Deputy Manager	Peer Reviewer (RP)
April 3, 2012	Rome, Italy	NA-SA's Governor / General Manager	26 th WANO Paris Centre General Assembly
April 9-27, 2012	Penly NPP, France	Atucha I NPP Training Manager	Peer Reviewer (TQ)
May 14 - June 01, 2012	Hunterston B NPP, UK	Embalse NPP I&C Engineer	Peer Reviewer (MA)
May 14 - June 01, 2012	Hunterston B NPP, UK	Embalse NPP Chemical Engineer	Peer Reviewer (CY)
May 21-24, 2012	Nantes, France	Atucha I NPP Industrial Protection Deputy Manager	Fire Protection Workshop
June 11-15, 2012	Embalse NPP, Argentina		Technical Support Mission on Operation Focus
June 18-22, 2012	HONG YAN HE NPP, China	Atucha II NPP Chief Supervisor	Reactor Operator Training TSM
June 28, 2012	Stockholm, Sweden	NA-SA's Governor / General Manager	53 th WANO Paris Centre Governing Board
September 3-7, 2012	Paluel NPP, France	Atucha I NPP QA Senior Engineer	Corrective action programme TSM
September 10-28, 2012	Sellafield, England	Atucha II NPP Chief Supervisor	Peer Reviewer (OP)
September 12-14, 2012	WANO PC	NA-SA's WIO	47 th WIOs' Meeting
September 13-27, 2012	Dukovany NPP (WANO MC) – Czech Republic	Embalse NPP Assistant to Plant Manager	Peer Reviewer (OA)
September 17 - Oct 5, 2012	Borssele NPP – Netherlands	Atucha I NPP Chief Supervisor	Peer Reviewer (OP)
October 8-11, 2012	Oropesa, Spain	EQ Electrical Engineer	SOER 2011-1 Transformer Reliability Workshop
October 8-12, 2012	Wylfa NPP, UK	Atucha I NPP I&C Manager	Embedment of Human Performances TSM
October 11, 2012	WANO PC	NA-SA's Governor / General Manager	27 th WANO Paris Centre Extraordinary General Assembly
October 11, 2012	WANO PC	NA-SA's Governor / General Manager	54 th WANO Paris Centre Governing Board
Oct 15 - Nov 02, 2012	Dungeness NPP – UK	Embalse NPP Senior Engineer	Peer Reviewer (ES)

PERIOD	HOST	PARTICIPANTS	ACTIVITY
October 18-19, 2012	NASA Headquarters		Seminar on Corporate Safety Culture
December 3-21, 2012	Atucha I NPP, Argentina		Peer Review
January 21 - February 8, 2013	Paluel NPP, France	Atucha II NPP Chief Supervisor	Peer Reviewer (OP)
January 21 - February 8, 2013	Doel NPP, Belgium	Embalse NPP Emergency Preparedness Supervisor	Peer Reviewer (EP)
February 15-17, 2013	WANO PC	NA-SA's WIO	48 th WIOs' Meeting
February 25 - March 15, 2013	Ringhals NPP, Sweden	Embalse NPP Technical Assistant for I & C Manager	Peer Reviewer (MA)
February 28, 2013	WANO PC	NA-SA's Governor / General Manager	28 th WANO Paris Centre General Assembly
February 28, 2013	WANO PC	NA-SA's Governor / General Manager	56 th WANO Paris Centre Governing Board
April 23, 2013	Antwerp, Belgium	Atucha II NPP, Senior Engineer	Doel Reactor Vessels Conference

In order to improve some particular areas, NA-SA requested during 2012 WANO's assistance through two Technical Support Missions and a Peer Review in CNAI held in December 2012. NA-SA personnel participated in three Technical Support Missions as well as eight Peer Reviews performed in other plants in the world.

Besides, it is foreseen the following activities to be performed during the rest of 2013:

- WANO Pre Start up Peer Review for CNA II.
- Technical Support missions: emergency preparedness and; fundamentals of system engineering (CNA I); lifting and rigging and; internal TSMs effectiveness (CNE) and; corporate nuclear oversight.
- WANO Peer Reviews, Technical Support Missions and Workshops: experts from NA-SA will participate in seven different missions.

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 Act N° 24.804, that regulates the nuclear activity and other activities, and Act N° 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 *Fourth* National Report to the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management (JCSSFMSRW) (2011) presents the following radioactive waste management policy:

- The radioactive wastes originated from all nuclear applications performed in the country, including wastes derived from the dismantling of related facilities, will be managed safely.
- *The allocation of responsibility for the development of radioactive waste management corresponds to the National State through CNEA having the generator the obligation of providing the necessary resources for such management.*
- The management of radioactive wastes will be performed safely, ensuring the protection and the rights of present and future generations and the environment.

- The Radioactive Waste National Management Program will be periodically reviewed and audited by the National Parliament.
- The establishment of a proper 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.
- *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 program for public communication program will be implemented.*

CNEA, as Responsible Institution for the application of the Radioactive Waste Management (Act Nº 25.018), establishes the acceptance criteria for Radioactive Waste conditioning and final disposal

3.19.9.2. SPENT FUEL MANAGEMENT POLICY

The following paragraphs excerpted from the *Fourth* National Report to the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management (2011) present the Spent Fuel Management Policy:

"In Argentina the Government exercises state ownership of special radioactive fission material contained in spent fuels from any origin: nuclear power plants and experimental, research and/or production reactors (Article 2, Act Nº 24.804). In this sense, the decision to reuse fissile material contained in spent fuels or not, 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, which must be operative by the year 2060 (Strategic Plan – Act Nº 25.018)."

3.19.9.3. SPENT FUEL TREATMENT AND STORAGE AT NPPs

3.19.9.3.1. Atucha 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.

The available positions in the pools will be used up by February 2015, and by that date a new option must be available for the spent fuel storage. Considering that date, NA-SA and CNEA have studied different alternatives for interim dry storage. *Therefore, an interim dry storage project has been developed.* Once it is built, spent fuels will be gradually transferred according to the decay time.

The interim dry storage project consists of a new building, where vertical underground silos will be located, constructed as an extension of the spent fuel storage building 1, located within the plant controlled area. The following is the present status of the project:

In January 2013 the interference settlement stage in the site of the new spent fuel storage building was finalized. The basic engineering, which is CNEA's responsibility, is still under execution.

Subsequently, the stage of civil works began with the construction of a net for sustaining soil, excavation and pit foundation and is currently on-going. The works responsibility was delegated to the Construction Unit of CNA II and Unit IV's Construction Group on the following topics:

- *Execution the detailed engineering Project.*
- *Execution of civil works.*
- *Execution the electromechanical erection.*
- *Commissioning.*

Despite this, the schedule is too tightened for that the facility be operational by February 2015. For this reason, and in order not to affect the operation of CNA I, the Construction Unit of CNA II is working on the following alternatives:

- a) Releasing the places currently occupied by internal components of the reactor located in the spent fuel storage pools. During the 2013 scheduled outage the plant began to release such places in such a way that would allow about four extra months of operation.*

- b) *Transfer spent fuel elements from the Unit I to the Unit II: By the end 2012 the conceptual engineering project to transfer 620 spent fuel elements from Unit I to Unit II was began with the firm AREVA-Transnuclear participation. At present, a negotiation is under way with the same company to carry out the detailed engineering and Preliminary Safety Report for this project.*

The licensing process of equipment and the spent fuel transfer operation will begin when be provided the above mentioned documentation. The fabrication of casks and other devices necessary for the transfer operation is foreseen to be available in by September 2014 in order to begin the process of sending spent fuels to the CNA II fuel storage pools.

The design's EOL of CNA I would be reached by the end of 2017, coincident with the completion of the spent fuel transfer to the CNA II fuel storage pools. At the same time, this solution serves as an alternative project until the spent fuels dry storage building is finalized.

3.19.9.3.2. Embalse NPP

As was informed in the report for the last convention (JCSSFMSRW) the CNE's spent fuel dry storage system (ASECQ) had till 2009 216 silos. *Currently the ASECQ system has 248 silos. The stored inventory at the end of 2012 was 99,900 fuel elements in 185 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. Atucha 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 60°C.

Each of four spent fuel storage pools has a 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 853 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.

3.19.9.4. RADIOACTIVE WASTE PRACTICES AT NPPs

Radioactive waste management at NPPs was described in detail in the *Fourth* National Report presented to the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management. The following is a brief summary:

In the case of low level liquid wastes generated from NPPs, the management is different at each plant on account of the different technologies used.

- Liquid radioactive wastes generated at CNA I during operation and maintenance activities are collected in tanks, characterized and concentrated by evaporation; concentrates as well as sludge from the clean-up of tanks are immobilized in cement matrixes and conditioned in 200 litre drums.
- 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. Such conditioned containers are stored temporarily at the facility.

Intermediate level radioactive solid wastes originated in the operation and maintenance activities of both NPP's, consist mainly of filters and spent ionic exchange resins. These intermediate level radioactive solid wastes are stored at the facilities of each NPP. All the storage facilities are located at the NPP sites for an interim storage.

At the Ezeiza Waste Management Area (AGE), there is an especially designed interim storage facility where non-conditioned wastes may be stored prior to their processing as well as conditioned waste packages awaiting their transport and/or final disposal.

Since 2008, following a regulatory requirement, the operator developed a program for the characterization of solid and liquid radioactive wastes generated at the NPP and begun its implementation. This has as an objective, among others, the building of a data base for the design of future repositories. The program includes training of personnel and developing of a capability for waste handling and characterization. The program involves all wastes produced since the beginning of operation. This characterization also includes solid non compactable wastes as well as contaminated structural elements.

3.19.9.4.1. CNE life extension: waste management program

The waste management program developed for the activities to be performed during the CNE refurbishment is based on the following considerations:

- *Regarding the waste generation, it is not expected that gaseous waste emissions exceed neither normal emissions, nor the liquid waste those of scheduled outages. The main radioactive waste corresponds to solid waste.*
- *Regarding the waste characterization, it has been estimated the radioactive inventory using samples, estimations, direct and indirect calculations in each case, estimated for a long term of 50 years.*
- *Only 15% of the total volume of waste removed during plant refurbishing outage corresponds to activated waste, while the large percentage corresponds to contaminated waste.*
- *All deposits containing fuels channels (re-tubing) and SG's tube bundles that will be replaced are designed so that their integrity will be maintained for, at least, 50 years.*

Regarding the re-tubing waste management, the CANDU Energy Inc. recommendations will be followed in the stages of treatment, conditioning and transfer. Such waste has no decontamination process, because it will be stored in vaults or containers once they are removed from the calandria.

For the waste coming from the SGs and moderator heat exchangers replacements, a pre-assembled concrete building with capacity for up to five SGs will be built. The SGs' pieces there installed will be, in themselves, a sealed source with a very low risk of contamination spread.

The rest of the waste, compactable, non-compactable and structural will be stored in the same place where now compactable wastes generated in the plant are stored, also new facilities are being foreseen.

3.19.9.5. RESEARCH AND DEVELOPMENTMENT ACTIVITIES

Some activities have been started in the past, and must be continued in the coming years in order to achieve the expected results. Others activities started during 2008 and 2009. *Besides, others activities are analysed so as to create new projects or to be included in existing projects and, the necessary costs, time and human resources are estimated, taking into account the appropriate capacities of CNEA and other organizations.*

The on-going activities of research and development are listed as follows:

- *Studies and the selection of radiochemistry techniques to a precise characterization of radioactive wastes and verification of the conditioned wastes quality.*
- *Development of equipment to verify the conditioned wastes quality.*
- *Development of a process for the treatment and conditioning of exhausted ion exchange resins, generated during the reactors operation which are stored in CNA I, CNE and AGE.*

- *Development of a process for decontamination/solidification of oils generated during the manufacture of fuel elements.*
- *Development of new materials for immobilization of low and medium level wastes.*
- *Studies on the behaviour in the long term of engineering barriers for their use in repositories for medium level radioactive wastes. This activity entails the development of new lasting concrete, their characterization and evaluation, through destructive and non-destructive assays as well as their behaviour in the long term as an engineering barrier for isolation of radioactive wastes in conditions of final disposal.*
- *Study of speed of corrosion of steel frameworks and transport parameters in armed concrete of intermediate level waste containers.*
- *Study to determine the corrosion rate of steel drums containing solid radioactive wastes (ion exchange resins and ashes coming from an incineration process) immobilized by cementation.*
- *Studies of corrosion of high level radioactive waste containers.*
- *Studies of differed breaks induced by hydrogen in the zircaloy cladding of the fuel in NPPs in operation and in spent fuels during the long term interim storage.*
- *The study of different compositions of ferrous phosphate glasses and the determination of the effect of the presence of uranium oxides, for the immobilization of high level radioactive wastes contained in the spent fuel of NPPs.*
- *Geologic environmental study fit for repository site for final disposal of low, intermediate and high level radioactive wastes.*
- *Modelling of water circulation in fractured crystalline rocky means.*
- *Modelling of water circulation in sedimentary rocky environment.*
- *Study of environmental characterization of sedimentary rocky environment of the AGE, whose knowledge will be also applied in the determination of the environmental base line of new possible sites of interest.*
- *The Geographic Information System continues and progress has been made.*
- *Digitalization of geological information of several country regions.*
- *Conceptual design of an interim dry storage system of the spent fuels from CNA I.*

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 ⁶⁰Co generation and the activity in operational wastes.

Further details were presented in the *Fourth* 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 and *2011*). Any further information can be obtained from the above mentioned National Reports that can be downloaded from:

http://www.cnea.gov.ar/politica_ambiental/seguridad_ambiental.php

3.19.10. COMPLIANCE WITH THE OBLIGATIONS IMPOSED BY THE CONVENTION

The information contained in this and other Articles demonstrates that Argentina complies with the obligations imposed in Article 19 of the Convention on Nuclear Safety.

CHAPTER 4

PLANNED ACTIVITIES TO IMPROVE SAFETY

The ARN and NA-SA identified a number of challenges during this reported period. Those challenges lead to actions with regards to both safety-related issues and regulatory issues in order to maintain and improve the safety level of the Argentine nuclear power plants during their remaining operating lives, as well as to improve the regulatory framework. In the following paragraphs, the intended measures are presented in a summary form.

4.1. REFURBISHMENT ACTIVITIES OF EMBALSE NPP

As were mentioned in section 3.6.5.1., NA-SA has developed a PLIM and PLEX program in order to maintain a high level of safety and plant performance during its long term operation (extended period of 25 – 30 year).

The Regulatory Body required to the Licensee, as a condition for licensing the CNE long term operation (LTO), to carry out a Periodic Safety Review (PSR), following the IAEA guidelines Safety Standards Series No. NS-G-2.1 (see section 3.14.3.3.). Besides, an update of the mandatory documentation including the Final Safety Report (FSAR) also was required.

Among the main CNE refurbishment activities could be mentioned the following:

- Re-tubing: the process consists, in the removal and replacement of the fuel channels, including the corresponding PTs (end fittings and others PT accessories); CTs; and feeders.*
- SGs replacement: includes the replacement of the cartridges (SG's tube plate and tube's bundle), the pressure vessel envelope, and the steam drum internals (primary moisture separators) for the four SGs.*
- Trip coverage's improvement of the reactor's shutdown systems (SDS # 1&2) for all original design basis accidents (DBA) and for some new events (BDBA).*
- ECCS's reliability improvement: automatic initiation even for very small LOCA, automatic switch from medium to low pressure injection, improve closure of medium pressure lines, extra lines for injection, relocation of sump level sensors to improve measurement, etc.*
- Reduce interface LOCA event frequency through ECCS line.*
- Protect main PHTS pumps by automatic trips.*
- Improve seismic capacity of EPS and EWS systems.*
- Improve the severe accident response, by removing hydrogen from the containment inner (PARs installation) and promote mixing within RB, and add an alternative heat removal, as well as to increase the calandria's venting for the case of a severe accident characterized by the moderator's heat sink failure.*
- Add an emergency filtered venting for RB.*

4.2. IMPROVEMENT TECHNICAL CAPACITY ON AGEING ASSESSMENT

The Argentina, as a State Member of IAEA, had been participating on extra budgetary program on 'International Generic Ageing Lessons Learned' (IGALL). The Argentinean's working group was integrated by ARN, NA-SA and CNEA and covered three areas of work: Mechanical; Electrical and I&C Components and; Civil Structures. The program objectives are to develop the "IGALL Document", which will provide:

- A roadmap to ageing mechanisms and effects based on research results and operational experience accumulated to date and;*
- A common internationally agreed basis on what constitutes an acceptable ageing management program for typical plant's SSCs, material, and environments which will be updated periodically with the latest activities on aging issues of the member states.*

The objective of the participation in the IGALL Program of ARN, NA-SA and CNEA is to increase their technical capacity in the ageing management, training their personnel to improve the monitoring and control tasks of the plants' ageing management programs as well as to provide insights in the identification of criteria and practices applicable to continued operation of the plants. In this context, NPP's LTO is intended as operation beyond the service life assumed in the design of plant.

An ad-hoc group was created in ARN to follow all control activities related to ageing of NPPs. In case of the Licensee, changes in the organization are under development to improve the ageing activities (see article 3.14.2.). Activities on this subject will continue in CNA I, CNA II and CNE following specific programs for each one including training on materials degradation.

4.3. IMPLEMENTATION OF THE IMPROVEMENTS ARISEN FROM STRESS TESTS

In response to the Fukushima Daiichi accident, among the efforts to strengthen the nuclear safety by achieving the objectives of the IAEA Action Plan and the NSC and to maximize the benefit of the lessons learned, the Regulatory Body required the Licensee to conduct a stress test in CNA I, CNA II and CNE in order to detect weaknesses to face in case of extreme events and to propose the corresponding improvements (see sections 1.4.1., 3.14.3.1.1., 3.14.3.1.2.1., 3.14.3.1.2.2., 3.14.3.1.2.3., 3.16.2.9., 3.17.2.3.3.1.1., 3.17.2.3.3.2.1., 3.17.2.3.3.1.2., 3.17.2.3.3.2.2., 3.17.2.3.3.1.3. and 3.17.2.3.3.2.3.).

As result of the stress tests carried out have been proposed numerous improvements and modifications to CNA I, CNA I and CNE. The main improvements and modifications, that have been foreseen to be implemented during the next three years, are listed below:

- *Alternative water sources (CNA I / CNA II / CNE).*
- *Passive auto-catalytic recombiners (PARs) (CNA I / CNA II / CNE).*
- *Venting filtered containment system (CNA I / CNA II / CNE).*
- *Alternative power sources –MDGs- (CNA I / CNA II / CNE).*
- *Re-evaluation of the risk of tornadoes (CNA I / CNA II / CNE).*
- *I & C improvements to provide information for severe accident management (beyond design basis accidents).(CNA I / CNA II / CNE).*
- *Fuel elements integrity assessment (CNA I / CNA II / CNE).*
- *Environmental monitor net around the plant (CNA I / CNA II / CNE).*
- *Source term verification and update (CNA I / CNA II / CNE).*
- *Safety seismic assessment (SSA) based on the corresponding updated seismic hazard (CNA I / CNA II).*
- *New hydrologic and hydraulic study (CNA I / CNA II).*
- *Installation of an additional (fourth) pump to the river Water Cooling Secured System (UK) (CNA I).*
- *Re-evaluation of the consequences of probable earthquakes on the existing dam located downstream (CNE).*
- *New Emergency Power Supply System (EPS) (CNA I).*
- *Electrical interconnection between normal bars of CNA I / CNA II.*
- *Assessment of the external electrical supply lines availability (CNA I / CNA II).*
- *Upgrade of the EPS system (CNE).*
- *Replacement of the Class III DGs (CNE).*
- *Extension of the batteries availability (CNA I / CNA II / CNE).*
- *Connection among the DGs of CNA I and CNA II (CNA I / CNA II).*
- *Modifications to the emergency water supply (EWS) system (CNE).*
- *Cooling of the RPV external side (CNA I / CNA II).*
- *Conditioning of a municipal emergency control centre far from the planning zone of the nuclear emergency (CNA I / CNA II / CNE).*

- *Emergency exercises extended in time (long term) (CNA I / CNA II / CNE).*
- *Relocation of the Emergency Control Internal Centre (CICE) (CNA I / CNA II).*
- *Modification and improvement of the Emergency Internal Control Centre (CICE) building (CNE).*
- *Improvement of the safety system trip parameters (CNE).*
- *Improvement of the ECCS' reliability (CNE).*
- *Secondary control room gateway reinforcement (CNE).*

4.4. INTERNAL TECHNICAL SUPPORT MISSIONS

The Safety Culture Strengthening Program (SCSP) is one of the activities that are being held with WANO support, which is currently developed in the following four areas: managers in the field, human error prevention tools, operating experience and emergency preparedness. Besides, it is foreseen to add the following topics: industrial safety, foreign material's exclusion (FME), fire protection and, ALARA practices.

The corresponding system to assess the SCSP's results has been developed by using of indicators, self-evaluation and polls.

In the above mentioned context, NA-SA has established an own program for crossed support missions between CNA I, CNA II and CNE, which are named Internal Technical Support Missions (ITSM), which is based on the methodology of the WANO's Technical Support Mission (TSM) . On this context, NA-SA requested a WANO TSM to assess and improve their ISTM, which was held at CNE in June 2013. WANO considered that this is an innovative and valuable tool with potential to be applied in other organizations with multiple sites.

4.5. USE OF OWN SPECIFIC PLANT SIMULATORS AS AN ADDITIONAL TOOL FOR PERSONNEL TRAINING / RETRAINING

The training / retraining of plant personnel belonging to CNA I, CNA II and CNE will be improved by using specific plant simulators according the following:

- *CNA I: A new interactive reduced scope windows basis graphic simulator was acquired to be added to the existing one as an additional tool for personnel training / retraining.*
- *CNA II: A full scope simulator and an interactive graphic simulator were commissioned to the enterprise TECNATOM and it was performed with NA-SA personnel participation. Both simulators have 120 plant systems incorporated. Models are foreseen to be updated with the latest engineering changes and commissioning data.*
- *CNE: A Full Scope Simulator developed by the enterprise L3-MAPPs was acquired. This simulator is similar to those installed in similar CANDU NPPs around the world and, will be the tool used for training and retraining of operating personnel on normal operation and accident conditions within the design basis during the refurbishment period and the second operating cycle (extended plant operation).*

On the other hand, ARN is asking technical assistance in the use of simulators for personnel licensing (for granting licenses and specific authorizations as well as its renewal).

ANNEXES

ANNEX I

CONCLUSIONS ABOUT ARGENTINA DURING THE FIFTH REVIEW MEETING ON THE CONVENTION ON NUCLEAR SAFETY

RAPORTEUR REPORT

I.1. SUMMARY

- Argentina has been a Contracting Party since 1994.
- There are currently two operating NPP's.
 - o Atucha I - 335 MWe PHWR.
 - o Embalse – 600 MWe HWR (CANDU).
- One new NPP is under construction.
 - o Atucha II – 693 MWe – update of Atucha I (Konvoy).
- One new reactor prototype is under planning – advanced.
 - o CAREM – 25 MWe – Argentine prototype at Atucha site.
- One new NPP under feasibility studies (1000 MWe).
- Atucha II was granted a Construction License in 1981 – special attention is being given to assurance of adequacy of design and impacts of the lengthy delays in the construction/ start-up process.
- Embalse is preparing for a refurbishment outage with plans to extend life another 25 to 30 years (plus output increase of 35 MWe).
- Nuclear NPPs represent 3.7% of installed capacity and generate 7.1% of the energy in the Country.
- Requests were made to extend outage intervals in Argentina NPP's from 12 months to 18 months.
 - o Proposals were reviewed against USNRC and Argentina standards for impact on increase to core damage frequency.
 - o ARN approved the proposal for Embalse.
 - o The proposal for Atucha is currently under review by the Regulatory Body.

I.2. HIGHLIGHTED REGULATORY CHANGES

- The Argentina Government has issued a plan indicating the policy on reuse of used fuel must be decided by 2030 and deep geological repository must be operational by 2060.
- The requirement for a PSR every 10 years (to meet requirements as stated in IAEA Safety Standard NS-G-2.10 “Periodic Safety Review of Nuclear Power Plants”) has been placed in NPP Operating Licenses.
- Regulatory documents are being updated to provide direction for life extension and new build including Standard Siting for Class 1 Nuclear Installations and supporting Regulatory Guides.
- It was pointed out that Argentina has a well-developed framework and routinely reviews new information like IAEA documents to ensure it is consistent with International expectations (continuous improvement).

I.3. SAFETY ACHIEVEMENT

- Effort to optimize radiological protection has had positive impact on worker dose:
 - o For example, the collective dose for replacement of 4 pressure tubes in 2006 was 13.68 mSv and this was reduced to 4.08 mSv in 2009.
 - o Pressure tube inspection and repositioning programs consumed 314.83 mSv in 2007, 216.27 mSv in 2008 and 178.40 mSv in 2009.
- PEER evaluations were conducted at Atucha in 2006 with follow-up in 2009 and at Embalse in 2007 with follow-up in 2009.
- Extensive inspections at Embalse have been completed as part of the Life Management Program. The results indicate Embalse NPP is safe to operate to the Refurbishment outage and has helped define refurbishment program scope.
- ARN has its own independent laboratory to do independent confirmatory analysis – the laboratory was re-certified under the ISO/IEC17025 program.
- Regulating Body training programs were certified.
- Budget for the Regulatory Body was increased from \$ 47M pesos in 2007 to \$ 129.5M pesos in 2010.
- Staff increased from 312 in 2007 to 364 in 2010 with a strong knowledge management program to support it – post graduate courses were certified to the ISO 9001-2008 standard.
- NASA developed and implemented an Environmental Management System that meets the requirements of ISO 14001.
- ARN developed and implemented a Quality Management System that is based on ISO 9001.2008.
- A computer based system for the prediction of post accident public dose and radiological dispersion has been implemented.
- Strong programs have been advanced for life extension and aging management and include elements like external hazard review, and safety upgrade programs derived from PSA and review against modern standards.
- Emergency Operating Procedures have been developed for low power and warm-up conditions.

I.4. GOOD PRACTICES

- Argentina has made a special focus on ensuring the requirements for a strong Design Authority are in place and have put in place some new review bodies as a result. They have made extensive studies of this aspect of their program including:
 - o Review against IAEA documentation.
 - o Over 23 events in the past 3 years involving many other Countries and IAEA support missions to advise on this important area.
 - o A large focus on documentation and organizational design to support the approach.

The importance placed on this area and the extensive review using international expertise to ensure the approach is appropriate is a good model and could be helpful to other countries – both for existing programs and particularly for those setting up programs with non-domestic suppliers and are thinking about Design Authority and Configuration Management approaches.
- Argentina has decided to install Passive Autocatalytic Hydrogen Re-combiners in its NPP's. Other countries have similar programs so this is a "highlight".
- It is clear that there is an effective use of PSA to inform refurbishment and design activities. The requirement to do so is a regulatory requirement in Argentina. This is a practice for some other countries but not all. Since this is done in a number of other countries it is reported as a highlight.

- There has been considerable work in knowledge management for the Regulatory Body including the certification of some post graduate training programs. Several other countries are reporting on knowledge management initiatives. This is recorded as a highlight.

I.5. CHALLENGES

- The spent fuel pool at Atucha I will be full by 2015. Plans are underway to design and construct dry fuel storage at the site.
- Adequacy of training for staff for the Regulatory Body and knowledge retention will continue to be a challenge with retirements and NPP program growth. The NPP Operators will have a similar challenge.
- There has been significant progress in the area of Human Performance Programs that have been introduced at NPP's. It was also reported that of 23 events reported 9 were dominantly as a result of human error and 8 were as a result of equipment problems. With growth, new people and new programming it will continue to be an area of challenge to prevent human error.
- The refurbishment and construction programs are a large expansion of the NPP program in Argentina. It will require continued attention to oversight of quality and external suppliers.

I.6. SUGGESTIONS

- Metrics to monitor effectiveness of the Design Authority function and follow-up assessment of effectiveness could help ensure the good work to prepare for effective implementation of Design Authority is effective.
- Continuing effort on reporting and trending of low level events and trending of indicators can help with further human performance improvements.

I.7. PLANNED MEASURES TO IMPROVE SAFETY

- Argentina has decided to install full scope training simulators at its NPP sites.
- External hazards assessments including flooding and a seismic re-evaluation are underway – to be considered in the design requirements for existing and future NPP's.
- Continuation of the program to implement Severe Accident Management at the NPP's is planned.
- Continued strong use of international reviews, IAEA Missions and OSART and WANO reviews is planned going forward.
- Plant safety upgrades are planned including:
 - o Installation of Passive Autocatalytic H₂ re-combiners.
 - o ECI strainers.
 - o Upgrade of Emergency Power system at Atucha I.
 - o Completion of CNA II Level 2 PSA.
 - o Replacement of Pressure Tubes, Feeder pipes and Steam Generators during Embalse refurbishment.
- Argentina is working on a Strategic Communication Plan which consolidates ongoing work in this area and includes elements like public education, conflict resolution, communication with local municipalities, public generally and neighbours.
- It is planned to further enhance the current strong relationships with other countries to support the large work program (like the relationship with the Senior CANDU Regulators).

I.8. INITIAL INSIGHTS / ISSUES FROM FUKUSHIMA DAIICHI

- Argentina reported that it has had a number of ongoing programs that are relevant to the event. Specifically:
 - o Implementation of SAMG.
 - o Installation of PARS.
 - o Re-evaluation of sites for external hazards and seismic margin.
- The event emphasized the importance of this ongoing work.
- The event also emphasizes the need to have an effective way to disseminate accurate information to public and other countries quickly. It was noted that this is particularly difficult today with the extensive media coverage that has become normal.
- The event emphasizes the need to be compliant with the Conventions we have that are relevant to this event.
- Once there is a consolidated package of information it will be possible to examine at a higher level of detail to gain lessons learned and evaluate opportunities to improve.
- The initiative to hold a meeting in June was seen as a positive step.

ANNEX II

ANSWER TO QUESTIONS OR COMMENTS

NATIONAL NUCLEAR SAFETY REPORT - 2010

Nº 1

CNS-REF.-ART.: 8

PAGE OF REPORT:

CHAPTER OF NAT. REPORT:

Argentina is commended for its comprehensive and detailed report.

Argentina appreciates the comment from Canada.

Nº 2

CNS-REF.-ART.: 8

PAGE OF REPORT: 39

CHAPTER OF NAT. REPORT: 3.8.2.5

The report states that “...New tasks to enhance the knowledge management initiative were identified for further action. They refer to the assessment of critical analysis of regulatory knowledge on specific key areas of the organization and the development of reliable indicators to measure the institutional intellectual capital.” Could elaborate on how knowledge management has been applied in practice to retain regulatory knowledge in critical areas and provide some examples where this was successfully implemented?

The implementation of the ARN policy on KM in the last three years has permitted rescuing and retaining specific regulatory knowledge in critical areas as could be seen in the next examples:

- 1) Since 2008, more than 1,700 documents (the equivalent of around 50,000 pages with relevant information) have been preserved in a specific database. Managing of this database through the institutional library is being analyzed.
- 2) Knowledge elicitation of 6 exiting experts was performed and the collected data was used for elaborating knowledge maps. Besides, a programme of seminars and conferences involving the exiting experts was implemented and this programme is now considered as part of the continued education of the ARN staff.
- 3) “Criticality” was one of the critical knowledge areas identified in the institution in the primary analysis. In 2008, the ARN had a single expert in that theme, who was going into retirement.
 - A series of counter-measurements were taken with the aim to ensure retrieval of such expert knowledge:
 - A retired expert was involved in a plan of internal training activities for designated ARN personnel. In parallel,
 - A senior staff physicist was sent abroad (New Mexico University; USA) for a specialized course on criticality safety and,
 - This senior professional is now coaching a couple of juniors.
- 4) The identification of critical knowledge areas is still an ongoing task. A specific procedure for determining the risk factor of knowledge loss across the whole organization is at the moment in a test period in some specific sectors of the institution.

Nº 3

CNS-REF.-ART.: 14

PAGE OF REPORT: 73

CHAPTER OF NAT. REPORT: 3.14.3.1.

The section provides very good overview of the deterministic assessment of CNA and CNE with regards to the End of Life (EOL) and Long Term Operation (LTO). Please discuss the expectations and/or specific criteria set by the Regulatory Body regarding the assessment of EOL and LTO.

1.- CNA I

The CNA I EOL was estimated in 32 full equivalent power years operation (EFPY). The CNA I RPV surveillance program results showed that it could operate safely until EOL.

However, ARN requested additional safety studies consisting in carry out more samples of irradiation, tests and new evaluations regarding neutronics and PTS. The results of these studies demonstrated that the RPV integrity is guaranteed until 1.5 EOL.

Regarding the CNA I LTO, the licensee has not still asked the corresponding authorization.

2.- CNE

The Nuclear Regulatory Authority (ARN) and Nucleoeléctrica Argentina S.A. (NA-SA) agreed according it is established in the "Licensing Bases Document" the general criteria applied to the assessment of Embalse NPP EOL and LTO.

The scope and contents of the deterministic studies will be defined between ARN and NASA according with the following information:

- A previous list of the studies that NASA will carry on
- A proposal of the scope established by ARN.
- A document that NA-SA should prepare with the hypothesis which are used for developing the PSA Level 1, including the fresh core condition cases.
- The acceptance criteria of these studies,

Up to now, because of the PLEX project delay, the deterministic studies list has not still been presented to ARN. So it is expected that NA-SA presents these studies and discuss with ARN the topics above mentioned during the current year.

However, regarding the most relevant safety related issues it could be mentioned the following ARN expectations:

2.1.- Pressure Tubes: NA-SA will continue demonstrating deterministically that safe operation can be sustained until EOL. In order to ensure safe operation, the criterion set by ARN comprises several aspects of PT integrity, such as, among others:

- No PT reaches blister formation susceptibility at any point of its length. Such assurance can be credited by avoiding PT / CT contact by garter springs repositioning as well as by verifying that equivalent hydrogen concentration in PTs would not reach the blister formation threshold (BFT) in PT / CT contact points before EOL by mean of scrape samples analysis.
- No PT is foreseen to run off-bearing before EOL due to elongation.
- LBB can be credited to be detected before a crack reaches critical length, until EOL.
- The periodic inspection programme is adequate to monitor diametral expansion and wall thinning to remain within design boundaries.

The necessary assessments to demonstrate that such criteria are met, are periodically required to NA-SA and, through independent assessments of the studies performed by NA-SA, ARN verifies the results adequacy.

Regarding LTO, it was decided to replace all the fuel channels by new ones; it means pressure tubes, calandria tubes and feeders, in a like-for-like approach during refurbishment outage.

2.2.- Feeders: As to ensure safe operation until EOL, NA-SA will continue assessing aspects such as:

- No feeder exhibits flaws that could lead to cracks (including body and welds).
- Wall thickness remains above critical thickness until EOL or, in cases where wall thinning is expected to go below critical thickness before EOL, a fitness-for-service assessment is performed to assure that the component will safely withstand operating conditions until EOL.

As was mentioned, regarding LTO, it was decided to replace all feeders by like-for-like new ones during refurbishment outage.

2.3.-Steam generators:

Due to Flow Accelerated Corrosion (FAC), the steam generator (SG) tube support plates (TSP) started to degrade producing SG tube damage due to fretting wear in the U – Bend. This has caused several unplanned outages due to leakages in tubes, as well as a progressive degradation and loss of function of TSP. Damages were observed in tubes due to fretting wear with scallop bars the U-Bend supports of tube bundles, known as "scallop bars" at tube bundles U– Bend.

This encouraged CNE to take corrective and mitigation actions concerning SG as part of the Life Management Program, being installed auxiliary supports known as anti-vibration bars (AVB's) which consists of three comb-shaped supports, for each SG, inserted in the U-bend of tube bundles. In addition, among others, the following corrective actions were implemented:

- Feed water chemical analysis through it was decided to change the amine used in AVT (All Volatile Treatment) treatment, in secondary circuit, in order to increase pH ;
- SG tubes plugging criterion was changed to a more conservative position and the inspection zone was extended;

- Water-lancing of the hot leg side of TSP, in order to clean “broached holes” and normalize secondary side flow;
- Program to follow-up the vibration conditions and preventive plugging due to fretting in U – Bend.

Considering the detected SGs degradation, ARN required an assessment on structural integrity of the components involved and; a verification of acceptance criteria established for tubes, support structures and SG internals in order to verify safe operation with an acceptable safety margin, being the main conclusions:

- The condition of the SG TSP was considered acceptable until EOL, both for normal operation and for transients caused by design basis accidents;
- SG tubes would remain stable without significant wear, without breaks for fatigue or damage to tubes due to interaction among them; and
- In main steam line break conditions, the initiation of fatigue cracks is not expected and it is concluded that the tubes will remain stable until EOL.

As a result of SG’s Life Assessment, it was concluded that SG’s Life Extension is not feasible due to TSP FAC degradation. Steam Drums will remain, while SG’s cartridges and tubes will be replaced. Also, it was determined that moisture separators are not in conditions to face LTO and will have to be replaced.

Regarding EOL, the TSPs status is still being followed. However, since 2007 there have been no failures in the SG tubes. Therefore, it is considered that the corrective actions taken have been effective.

In addition, due to the studies performed have shown that the SGs could operate safely until EOL (210,000 EFPH), in order to consider authorizing the LTO for 15,000 EFPH beyond EOL, ARN has requested to NASA to assess the SGs condition corresponding to 225,000 EFPH in order to demonstrate that the SGs would operate safely for this additional period.

The above mentioned regulatory requirement includes an analysis of the TSPs degradation rate, the structural integrity considering the SGs current state and the consequences of seismic occurrences and other events that could cause a sudden secondary circuit depressurization.

Nº 4

CNS-REF.-ART.: 14

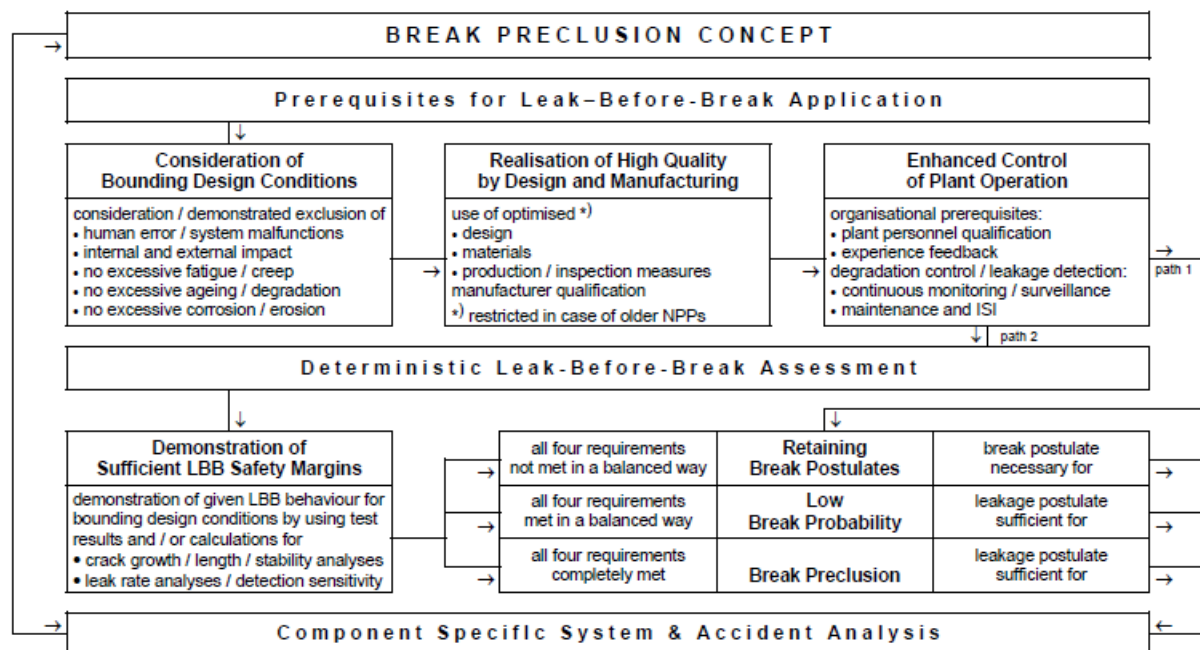
PAGE OF REPORT: 112

CHAPTER OF NAT. REPORT: 3.18.3.3.2.4.

This section discusses the evaluation of the break preclusion concept. Please summarize the Regulatory Body position on the application of the leak-before-break concept to ageing safety significant systems, structures and components.

In relation to the LBB demonstration, there is no a specific related Argentinean standard. The regulatory position is defined case by case following the state of the art and the international validated practices or rules, as for example Standard Review Plan 3.6.3.

In the particular case of Atucha II, the Break Preclusion Concept applied to the design, requires the fulfilment of the prerequisites for the Leak- Before- Break application, as stated in the below diagram.



As part of the bounding design condition it is necessary to demonstrate the exclusion of excessive ageing / degradation in the intended piping. For Atucha II and in order to assure that no ageing / degradation arose from the long storage period, both inspection of finished and semi-finished products, and storage procedures and conditions were performed on a spot check basis. These activities were considered as sufficient by the Regulatory Body in order to assure that the material properties are still according to the specification and so, the LBB behaviour could be applied.

Nº 5

CNS-REF.-ART.: 14

PAGE OF REPORT: 73

CHAPTER OF NAT. REPORT: 3.14.3.1.2.

This section discusses in detail the accident management and severe accident management program. In Annex V we noticed that the CNE NPP considered hypothetical multiple steam generator tube breaks in the Final Safety Report. Do the safety analyses of the CNA NPP consider multiple steam generator tube failures also? Are there significant safety related differences regarding the regulatory requirements, estimated frequencies, and consequences of such event(s) between the two plants?

In the case of CNA NPPs, the multiple steam generator (SG) tube failures is considered in the Probabilistic Safety Assessment (PSA) and Severe Accident Management Program (SAMP).

In relation with the safety related to differences regarding the regulatory requirements for CNE, the hypothetical multiple steam generator tube breaks is considered in the Plant Life Extension Program which include, as a task of the second phase of this program, the Steam Generators Replacement. This event for CNA, in addition to other requirements, is considered as part of the safety analysis included in the Periodic Safety Review.

Regarding the consequences of such event(s), they are similar for both NPP's because for some transients, the SG's of CNE perform similar functions that the Second Heat Sink in the case of CNA.

Nº 6

CNS-REF.-ART.: 2

PAGE OF REPORT: 5

CHAPTER OF NAT. REPORT: 2.1.2.

As a result of ARN's decision to require Licensees to develop a Severe Accident Management Program (SAMP), have design changes been implemented or planned to address SAMP, and if so could they briefly be described?

Atucha I NPP: As mentioned in the Argentine Report (see page 74), the basic engineering for design changes needed to implement the inventory replacement strategy using the volume and pressurizing control System was completed. Nevertheless, some engineering tasks focused on avoiding the heavy and light water mixing is under study by the plant engineering group. The engineering tasks in order to implement the recovery of the Second Heat Sink System (SHS) capability by injecting water to SHS make up tank is being analyzed by the plant engineering group.

As mentioned in the Argentine Report (See page 7), the Licensee is defining the insertion of the SAM team in the Internal Committee of Emergency Control structure. Also, a proposal for the Severe Accident Guides content is under discussion. Nevertheless, any change in the Licensee organization should be approved by the Regulatory Body before being implemented.

Also in the page 7 is written that a new version of the methodology to convert the strategies to Instructions and Guidelines was issued. The strategies from SAMP will be included into the Operation Manual as Instruction for SAM preventive domain (Instructions P) or under the set of Guidelines for SAM mitigating domain (Guidelines M).

Embalse NPP: All the foreseen design changes related to the SAMP will be implemented during the CNE Refurbishment process. There are two main aspects:

- Implementation of design improvements in order to deal with severe accident conditions. The improvements in the CNE design, related to severe accidents have two sources: the design changes recommended by Level 2 PSA results, and those recommended from the experience in others CANDU plants. Both sources have been considered by NA-SA.
- Development and implementation, with the support of the designer, of Severe Accident Management Guidelines for CNE using the generic guidelines for CANDU NPPs.

Nº 7

CNS-REF.-ART.: INTRODUCTION

PAGE OF REPORT: 2

CHAPTER OF NAT. REPORT: 1.3.

It is written that "CNE life extension project will extend plant life for another 25 year of full operation". Can Argentina clarify what standards have been used for safety assessment beyond design life?

As it is mentioned in the Argentinean National Report, "it is expected that the CNE Life Extension Project will extend the plant life for another 25 years of full power operation".

The extended operational period will depend of the final result of the safety evaluation that are on going according it is informed in section 3.14.3.1.3. of the Report.

The safety assessment beyond design life of Embalse Nuclear Power Plant is based on international recommendations. As an example, the following documents could be mentioned:

- 1) IAEA-Safety Standard Series - Periodic Safety Review on Nuclear Power Plants, Nº NS-G-2.1.0.
- 2) Canadian Regulatory Document RD-360 (Canadian Nuclear Safety Commission) Life Extension of Nuclear Power Plant.

These documents cover the safety factors that the plant has to consider, aimed to fulfill the requirements to extend the Plant Life Operation.

Nº 8

CNS-REF.-ART.: 2

PAGE OF REPORT: 11

CHAPTER OF NAT. REPORT: 2.2.3.

What design improvements have been identified / incorporated in order to minimize the consequences as a result of severe accident analyses?

Embalse NPP: All the foreseen design changes related to the SAMP will be implemented during the CNE Refurbishment process.

The design improvements to be implemented come from two sources: the design changes recommended by from the experience in others CANDU plants, and from the Level 2 PSA CANDU results.

The following design improvements are under consideration to be implemented:

Description of the design change	Reason for the change
Addition of passive auto-catalytic recombiners in the reactor building.	Remove hydrogen from the containment atmosphere. Promote mixing of the atmosphere within the containment to avoid the formation of regions with combustible gases.
Addition of a make-up line for the calandria vault, from outside the reactor building.	Ensure an alternative long term heat sink, for the case of a severe accident with loss of moderator heat sink.
Addition of a rupture disk to the calandria vault.	Increase the calandria vault venting capacity, to prevent pressure from rising above the design value after a severe accident with loss of moderator heat sink.
Addition of a containment filtered venting system.	Avoid a possible challenge to the containment due to over-pressurization. Reduce the driving force for the release of fission products outside the containment. Improve containment environmental conditions by removing radioactive products.

Additionally CNE is currently working in the development and implementation, with the support of the designer, of Severe Accident Management Guidelines using the generic guidelines for CANDU NPPs.

Nº 9

CNS-REF.-ART.: 2

PAGE OF REPORT: 10

CHAPTER OF NAT. REPORT: 2.2.3.

Updated information on the implementation of the PLIM and PLEX program at CNE would be welcomed during the forthcoming meeting in particular an “implementation priorities”.

The CNE Life Extension Project was divided into three phases. At present NA-SA continues working in the activities mentioned in PLEX phase 1, and the retubing and SG related tasks (initiated in middle 2010) corresponding to phase 2 which essentially requires the specification and purchase of replacements for reactor components, the planning of Phase 3 and the development of purchase and design documentation. In short, the development of all the appropriate actions to the completion of Phase 3 which requires a different organizational structure as that for Phase 1, regarding that both the actions and objectives of the project have changed from analysis and feasibility studies (Phase 1) to the development of a great amount of documentation, management and execution of large contracts, carrying out significant activities of temporary and permanent imports, planning, etc.

Phase 1 activities are contained in two reports already issued: “Ageing Assessments of Systems, Structures and Components” - which includes the definition of implementation priorities - and, “Safety Assessment Final Report”.

The company CONUAR is being qualified as a national supplier of the reactor components and CNEA laboratories for the implementation of metallurgical tests during the development of prototype components by CONUAR and during the manufacturing process.

The contract for the manufacture of the 4 new steam generator cartridges was signed with the company Industrias Metalúrgicas Pescarmona SAICyF (IMPESA), and the contract for technical assistance during the manufacturing process was signed with AECL.

AECL submitted the final report on the geological research conducted in the CNE site area, and a contract was signed with AECL to generate the floor response spectra (FRS) for all the buildings that are relevant to the PSA-based Seismic Margin as well as for the design and acquisition of equipment associated with the Embalse life extension.

A contract for the provision of a replica simulator of the Embalse control room was signed with the company L-3 Mapps.

Under a co-operation agreement between NA-SA and Hydro Québec, a contract for supply the documentation prepared for the accomplishment of the Environmental Qualification Standard for Nuclear Plants Equipment (CSA N290.13) was signed.

In September 2010, a Memorandum of Understanding on Co-operation in Nuclear Energy between the Argentina and Korea was signed aimed to analyze the possibility of the involvement of KEPCO (Korea Electric Power Corporation) / KHNP (Korea Hydro and Nuclear Power Co. Ltd.) in the development of the Fourth Nuclear Power Plant and the Embalse Life Extension projects.

Nº 10

CNS-REF.-ART.: 12

PAGE OF REPORT: 60

CHAPTER OF NAT. REPORT: 3.12.1.2.

What is the trend of precursors of events on NPP?

The groups responsible for the internal operating experience in each plant have a specific procedure and a data base for plant events. In the late revision of these procedures (2008) the treatment for precursors has been explicitly included with methodologies for their analysis which follow WANO guides and recommendations.

For each precursor identified (as well as in general for events) a report is produced. This report is sent to the sector involved (which can provide a solution) and it is incorporated to the data base. Periodically the plants OPEX groups elaborate a summary report which identify the areas with more findings. These areas are selected for improvement implementation. In that sense trends have to be taken into account.

Even when Human Performance Improving program begun in 2008 including the study of precursors, we are still in the process of data evaluation from 2008 to 2010 and it is not yet possible to withdraw conclusions about trends.

Nº 11

CNS-REF.-ART.: 12

PAGE OF REPORT: 1

CHAPTER OF NAT. REPORT: 1.1.

The report states that feasibility studies are in progress for a fourth NPP. Can you provide additional information regarding any schedules for decisions regarding this plant?

The feasibility Studies have been completed and the decision for the fourth NPP has been approved by Law 26566. During 2010 different potential suppliers have been prequalified. Current schedule is to issue the request for proposal for the fourth NPP by the end of the first semester of 2011.

Nº 12

CNS-REF.-ART.: 2

PAGE OF REPORT: 7

CHAPTER OF NAT. REPORT: 2.2.1.

The report states that a test facility is under construction that would verify the modifications to the Fast Boron Injection System (FBIS).

(1) Please provide an update of the status of the test facility.

(2) Clarify if tests have been performed already. If so, will changes to the FBIS be needed?

The facility is under construction. The mechanical erection was finished at the end of February 2011, and the electrical and I&C by March 15th. All the procedures for the test are already performed. It is foreseen to perform the experiments by middle of April 2011.

Nº 13

CNS-REF.-ART.: 6

PAGE OF REPORT: 19

CHAPTER OF NAT. REPORT: 3.6.4.1.3.

This section discusses briefly the technical basis for conducting component inspection programs during both programmed outages and planned shutdowns but does not offer any legislative or regulatory requirements that are used to initiate these inspections. Could you elaborate on the regulatory requirements used by Argentina to qualify non-destructive inspection techniques (testing methods), tools and personnel performance?

In Argentina, the In Service Inspection (ISI) Program is part of the mandatory documentation that the utility must provide to the Regulatory Body in order to support the Operating Licence. This program is normally based on the specific plant design, the engineering judgement and focussed in managing the foreseen degradation mechanisms according to the experience. It is up to the utility the definition of the appropriated program, which must contain different aspects like, selection of systems, components and structures to be inspected, determination of categories or inspection levels, inspection frequencies and the different inspection techniques to be applied. Once this information is submitted, ARN assess the adequacy of the inspection plan for the specific plant.

Normally the utility follows the requirements stated in ASME – Section XI Code, including or not the performance demonstration and EPRI guidelines revision 7 for the eddy current tests but this is not constrained to any specific regulatory requirements.

The ISI program establishes that the inspection personnel must be qualified in accordance with appropriate standards (IRAM standards) and recognized for each technique to be applied. The certification activity of the personnel qualification is administered by a Certification Body assisted by authorized rating agencies. On the other hand, the procedure applied states that "the person responsible for performing, monitoring and coordination of the scheduled inspection tasks must select the qualified and trained personnel to carry out the required inspection tasks. The qualification of the inspection personnel is verified by the plant quality control, and audited by the Regulatory Body.

Nº 14

CNS-REF.-ART.: 7

PAGE OF REPORT: 24

CHAPTER OF NAT. REPORT: 3.7.2.2.

It is reported that "to complete the harmonization with IAEA Standards some modifications may be required." What documents do you need to modify to adapt to the IAEA Standards?

The Argentine regulatory standards modification process requires several review steps, including consultation with stakeholders.

Moreover, Argentina is actively involved in all IAEA nuclear safety standards committees (SSC, NUSSC, RASSC, WASSC and TRANSSC).

Argentinean representatives in such committees report to the ARN Standards Committee about the relevant changes carried out in the IAEA safety standards who immediately began the corresponding harmonization process. Besides, as a manner to apply them ARN could issue specific requirements to the Licensee covering the time delayed until such changes are included in the corresponding Argentine Standard.

The paragraph § 3.7.2.2. in our National Report indicates that "as a result of the comparison of ARN Standards vs. IAEA Standards some improvement possibilities were identified in some non essential aspects of the ARN standards".

Most of the referred modifications that may be required to complete harmonization of ARN Standards with IAEA Standards, deal with the incorporation to the set of ARN standards and guides of some radiological safety criteria and safety requirements already imposed by the ARN to the licensees; i.e.: exemption levels and clearance levels for regulatory purposes, extension of the scope of some regulatory standards (i.e. decommissioning, quality systems) from NPP to all type of nuclear installations, severe accidents management in NPP, and emergency preparedness and management.

Nº 15

CNS-REF.-ART.: 7

PAGE OF REPORT: 27

CHAPTER OF NAT. REPORT: 3.7.3.2.2.

The license validity period (operating license) was limited to 10 years. After some experience does the regulator – ARN is considering to extend this period for e.g. 30 years maintaining a requirement for a formal PSR each 10 years?

The Regulatory Body (ARN) is not considering to extend the validity period of the NPP's Operating Licenses. The mentioned validity period is 10 years and it is required a PSR for its renewal.

The first operating license is issued by ARN after verify that the design is adequate for the plant lifetime (32 FPY for Atucha I NPP and 25 FPY for Embalse NPP). Therefore, after be granted the first plant Operating License, it is possible its renewal each 10 years within the plant lifetime, after PSR approval by the Regulatory Body.

Nº 16

CNS-REF.-ART.: 7

PAGE OF REPORT: 28

CHAPTER OF NAT. REPORT: 3.7.3.3.

Are the inspections planned e.g. by means of "Annual inspection plan" or are they performed casually – when abnormal event occurred?

As was informed in art. 3.7.3.3. of the Argentinean National Report, there are two different kind of regulatory inspections:

- Routine inspections are carried out by resident inspectors and other ARN personnel, and planned according to a Specific Procedure. Their purpose is to verify that the Licensee complies with limits and conditions of operation established in the mandatory documentation.
- Special 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. They have several purposes, e.g. to control preventive maintenance tasks during a NPP's programmed shutdown.

Nº 17

CNS-REF.-ART.: 7

PAGE OF REPORT: 28

CHAPTER OF NAT. REPORT: 3.7.3.4.

Are the corrective action, recommendation or good practices discussed with licensee and include in the document demands the licensee to carry out them in a certain time period with emphasize to prevention?

As a routine practice the Regulator maintains a close interaction with the Licensee by means of periodic meetings where, among other issues, are discussed those of regulatory concern which includes the application of recommendations or good practices arose from exchange of information, technical documents or lesson learnt from domestic or international sources focussed on emphasizing prevention.

As a result of those meetings, in some cases the Licensee decides by your own to carry out a corrective action. In other cases, the Regulator issues the corresponding regulatory requirement, if necessary.

Nº 18

CNS-REF.-ART.: 7

PAGE OF REPORT: 28

CHAPTER OF NAT. REPORT: 3.7.3.3.

What is the difference between inspections and audits? Are audits directed more to the field of the organization, process aspects and documentation? Inspections are directed more to the field of the technical and technological aspects?

Although it is correct that "audits are directed more to the field of the organization, process aspects and documentation", the main difference between inspections and audits is related to the

methodologies applied to perform them. For both of them, the objective evidence is compared against the acceptance criteria.

In case of routine or special inspections, the scope is more restricted; they could be not announced; carried out by only one inspector, and the inspector internal report is managed by ARN in order to take actions.

In case of audits, a plan (including: objective, scope, time table, etc) is always announced to the licensee and usually there is a previous meeting to prepare the audit. It is carried out by a team, who revised an important amount of information (some times historical) compiling objective evidence according to a sample criterion. Finally an audit report is issued and sent to the licensee.

Both inspections and audits are complementary regulatory tools and, in some cases, audits are planned as a consequence of the inspections results.

Nº 19

CNS-REF.-ART.: 7

PAGE OF REPORT: 27

CHAPTER OF NAT. REPORT: 3.7.3.2.3.

It is mentioned in the Report that the Regulatory standards set a system of issuing permits for personnel where along with the issue of individual licenses for personnel to perform certain functions there are also some specific authorizations to the licensed personnel for performance of specific functions at NPP. This system of specific authorizations seems to be over-regulated, which could negatively impact the compliance with the principle of ultimate responsibility of the operating organization. Could you please provide some additional comments to the above question?

The individual license (IL) and the specific authorization (SA), are two kinds of conceptually different and complementary documents which are required for holding a licensable position in a given NPP.

- 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.

The plant personnel who apply for a NPP licensable position must be proposed and trained by the Licensee in agreement with the Regulatory Body according to what is established in the Argentinean Regulatory Standards.

On the other hand, regarding the compliance of the principle of ultimate responsibility of the Licensee, the Operating License and the Regulatory Standard AR 3.9.1. establishes the following:

- The Licensee is the responsible organization for the nuclear and radiological safety; safeguards and plant security. Besides, the Licensee has to monitor that the plant is operating in a safe condition and has to give the support to the plant manager needed for fulfil its responsibilities;
- The plant manager, who is designated by the Licensee, is the direct responsible for the nuclear and radiological safety; safeguards and plant security ;
- The responsibilities of the Licensee and the plant manager can not be delegated.

Nº 20

CNS-REF.-ART.: 7

PAGE OF REPORT: 24

CHAPTER OF NAT. REPORT: 3.7.2.2.

Harmonization with IAEA Standards. It is stated that: "it is possible to conclude that Argentine standards are completely consistent with IAEA's corresponding documents, taking into account that ARN has mainly adopted a performance criterion". Please describe in more detail the performance criteria used and the comparison made with IAEA Safety Standards.

According it was mentioned in the section 3.7.2.3.1. of the National Report, aimed to clarify this point, it could be mentioned that the performance criterion philosophy is of a non prescriptive nature but related with the compliance of safety objectives. It means that the main concept of such criterion can be summarized by requiring to Licensees "what to do" instead of "how to do". As an example, see the Support Document Argentinean Standard 3.2.1."General Safety Criteria for Design".

As a result of the comparison between the ARN Regulatory Standards and the IAEA Safety Standards, it was identified the need to include to the ARN Standards and Guides some radiological safety criteria and safety requirements already required by the ARN to the Licensees; i.e.: exemption levels and clearance levels for regulatory purposes, extension of the scope of some regulatory standards (i.e. decommissioning, quality systems) from NPP to all type of nuclear installations, severe accidents management in NPP and emergency preparedness and management.

Nº 21

CNS-REF.-ART.: 7

PAGE OF REPORT: 26

CHAPTER OF NAT. REPORT: 3.7.3.2.2.

The report states that “SAR of NPPs must be updated each time that a plant design modification is performed and once every five (5) years”.

Does this mean that a reissue of the whole SAR occurs at each plant modification or that an amendment is produced for the parts of the SAR affected? Is a full SAR issued at the five year update, thereby integrating all design changes into a new assessment?

According to what is established in the Regulatory Standard AR 3.9.1., each time that a safety related plant design modification is performed the SAR must be updated by means of an amendment. Nowadays it is being revised/discussed this request considering a 10 years period to require the utilities to present, together with a PSR, a SAR updated version.

Nº 22

CNS-REF.-ART.: 7

PAGE OF REPORT: 29

CHAPTER OF NAT. REPORT: 3.7.4.

The sanctions regime described appears to prescribe sanctions only against the organization. Are any sanctions available against people?

The sanctions regime “Sanctions Regime for Nuclear Power Plants” also imposes graded sanctions against people who are licensed by the Nuclear Regulatory Authority in case of non compliance with the Mandatory Documentation.

Nº 23

CNS-REF.-ART.: 8

PAGE OF REPORT: 32 / 33

CHAPTER OF NAT. REPORT: 3.8.2.1.1.

Argentina may please provide whether areas i.e. Accident analysis, plant maintenance and surveillance and emergency preparedness are covered in training program.

The training programme for ARN staff in the areas of accident analysis, plant maintenance and surveillance and emergency preparedness have different approaches based on the national experience in those themes:

- Basic aspects of accidental analysis, plant maintenance and surveillance, emergency preparedness and other subjects related to radiation protection and safety are addressed in the Post-Graduate Courses of Radiological Protection and Nuclear Safety. These courses are run annually by the ARN for both, domestic and regional needs, the latter, in its role of IAEA Regional Training Center and they were established more than thirty years ago.
- Regarding accident analysis, ARN also sends abroad senior and semi-senior staff to participate in different courses, seminars and workshops, for example, to Germany, Canada, USA.
- ARN Emergency and Preparedness Unit has developed actions to keep up to date the staff response capability: In emergency preparedness where ARN has achieved a high level of experience, the specific sector organizes quarterly training activities for continual education of its own personnel. These activities are complemented with specific and periodic field rehearsal.
- An important mechanism of training of all the parties involved in the system is the participation in the planning, coordination and implementation of emergency drills. That involves not only the ARN designated staff (i.e., personnel specifically assigned to the E&P Unit, but many other regulatory personnel as well as staff from the support units, such as institutional communication.

- In the area of maintenance and surveillance, specific and advanced training is mainly organized through “on the job training” after the junior staff has completed ARN Post-Graduate courses. For instance, ARN's junior staff is trained in the field (NPP's) by experienced inspectors.
- Participation in international and regional forums, like the FORO, NERS, Candu Senior Regulators and others are also good opportunities to young professionals, that together with senior experts could be benefited by the exchange of information, lessons learnt and other experiences.

Nº 24

CNS-REF.-ART.: 8

PAGE OF REPORT: 38

CHAPTER OF NAT. REPORT: 3.8.2.3.5.

Could you please provide some examples of opportunities for improvement resulting from the self-assessments performed by ARN and referenced in the report in section 3.8.2.3.5., Measurement, Assessment and Improvement?

Examples of opportunities for improvement resulting from self-assessment are the following:

- Flow charts included in the documented procedures.
- Elaboration of procedures by meetings.
- Optimization of effectiveness indicators.
- Infrastructure improvements.
- Communication and relationship between areas and processes.

Nº 25

CNS-REF.-ART.: 8

PAGE OF REPORT: 39

CHAPTER OF NAT. REPORT: 3.8.3.

It is stated in the Report that the law sets annual fees for the licensees of the operating NPPs to fund the activity of the regulatory body. However, the order of this fee collection and its amount are not disclosed. Please, provide information regarding this issue to be sure that the financial independence of the regulatory body is not impaired.

The most important aspect is the independence of the regulatory body on the safety related decision making. As a consequence of this aspect the financial independence is sure.

The regulatory body depends on National Presidency and the NPP depends on the Minister of Energy. As it is stated in the Report, ARN annually issues a budget proposal with a detailed income provisions including regulatory fees, that for 2010 it was about 40% of the 2010 annual budget.

ARN ask to Economy Minister for annual budget considering the regulatory fees annual income.

Nº 26

CNS-REF.-ART.: 8

PAGE OF REPORT: 39

CHAPTER OF NAT. REPORT: 3.8.3.

There is a significant increase in the 2010 budget. What is the ratio (or total amount) to the overall budget for external technical support (e.g. independent assessment, R&D, etc.)? Is this a part of “services”?

The ratio to the overall 2010 budget for external technical support is 36%. This is a part of "services" (46%).

Nº 27

CNS-REF.-ART.: 8

PAGE OF REPORT: 34

CHAPTER OF NAT. REPORT: 3.8.2.1.3.

ARN has formed an ad hoc organization for commissioning of CNA II. Is this organization an integral part of ARN or is it an external “contracted” organization (fig. 3.8.1.). Does this organization controlling also the suppliers (e.t. their production, quality)?

In 2005 an “ad hoc” Division responsible for CNA II licensing was created. This Division was an integral part of ARN and all its staff belonged to ARN. Although some external technical services were

required during the licensing process, the external advisors or support organizations did not integrate this division.

In relation with the control of suppliers, this Division was responsible for all the inspections and assessment tasks related to CNA II licensing, including the verification of the suppliers qualification.

In 2010, as was explained in section 3.8.2. of the National Report, a new Control and Licensing of Nuclear Reactors Division was included in the ARN organization structure.

Nº 28

CNS-REF.-ART.: 8

PAGE OF REPORT: 30 / 31

CHAPTER OF NAT. REPORT: 3.8.2.

Section 3.8.1. describes the functions and competence of the Regulatory Body in Argentina including the approval of radioactive waste transport procedures, in particular, irradiated fuel elements, set by CNEA as a center producing radioactive waste. In this way ARN becomes the participant of these activities, which is, in our opinion, unacceptable for a regulatory body. In our vision, everything necessary to ensure the safety of transportation shall be set by the regulatory body in the corresponding regulations, e.g. standards, in accordance with which the work performer develops specific procedures to be followed under surveillance of the regulatory body. Could you please give us your comments regarding this issue?

In our opinion a misunderstanding between the concepts of transference and transport of spent fuel (SF) or radioactive waste (RW) appears in this question.

First we want to emphasize that in Argentina SF are not considered RW. The only transport of SF in Argentina is performed in the site of each NPP.

The transport of RW or SF, shall be accomplished with the National Standard for transport of radioactive material (that has adopted the IAEA TS-R-1).

The procedure for the transference between the SF wet storage to the SF dry storage shall be authorized by the ARN. In this case ARN acts also supervising the transference for safeguards reasons.

Nº 29

CNS-REF.-ART.: 8

PAGE OF REPORT: 32

CHAPTER OF NAT. REPORT: 3.8.2.1.

Taking account of the IAEA Safety Requirements for Governmental Legal and Regulatory Framework for Safety (GS-R-1 Part 1) regarding the issues of competences and training and development of the regulatory body, does ARN have a competence management system, within its overall management system, to deal with: systematically assessing competence needs, in the near term and longer term future, and planning and delivering training and other elements of competence development?

The ARN is in the process of implementation of a competence management system.

At present, the personnel general competences have been elaborated and they are now being reviewed.

Besides, all the job profiles are being updated in order to adequate them to such competence system to be linked to the personnel competences development.

The above mentioned job profiles will allow us to improve the existing training plans in the near term and longer future term.

Nº 30

CNS-REF.-ART.: 8

PAGE OF REPORT: 38

CHAPTER OF NAT. REPORT: 3.8.2.3.

Please also see 3.8.2.4. Communications with the stakeholders Page 38 and 4.2. Strategic Communications Plan Page 133

These sections refer to surveys of licensees and suppliers, and providing information to the public respectively.

Since communication is a two way process, Argentina is requested to clarify whether there are other stakeholder groups, other than licensees and suppliers, also surveyed for their opinion

on ARN and, if so, what are the questions asked of them, and how often are surveys made and with what size of sample? Are they carried out by an independent organization?

As expressed in the National Nuclear Safety Report 2010, the ARN designed, approved and undertook a Communication Strategic Plan by mid 2010. The particularity of this document is its modifiable nature as its implementation progresses. Its main objectives are to improve communication with both internal (1) and external stakeholders, and to strengthen ARN's institutional image.

According to ARN's communication strategy, the first step taken has been an internal survey to our staff. All questions have been oriented to identify which communicational profile the personnel consider ARN should have. Once processed this information, the result will become the foundations for developing ARN's institutional image.

On the other hand, surveys to users have been made with the aim of obtaining information regarding their perception of our institution, and also about the relationship held with ARN.

The next measure will be to unify and analyze all the data collected so as it can be used in future communication actions. Among other tasks, opinion polls addressed to area residents in the vicinity of power plants sites and other points of interest are planned.

It is our intention to carry out the mentioned studies initially with a small sample to later extend the application field, when needed. The objective of this work is to have a first approach of the public's perception about our institution, with the aim of working on our image and the bond with different stakeholders.

(1) Any ARN's officer is understood as an internal stakeholder

Nº 31

CNS-REF.-ART.: 9

PAGE OF REPORT: 45

CHAPTER OF NAT. REPORT: 3.9.2.

The Report indicates in Section 3.9.2. states that "The Licensee shall submit to the Regulatory Body the technical documents needed to evaluate the safety of the NPP which the operating licence is applying for." Presumably details of these requirements are contained in Regulatory Standard AR 0.0.1. and AR 10.1.1.; could you elaborate further on the reporting requirements imposed by the Regulatory Body in response to events or situations at operating NPPs, particularly with regards to those related to the degradation of ageing pressure boundaries.

The reporting requirements related to the degradation of ageing pressure boundaries is considered in a general way in the reporting requirements imposed by the Regulatory Body in response to events or situations at operating NPPs. Among others, it can be mentioned the following regulatory requirements contained in the plant Mandatory Documentation:

- An event analytical report must be submitted to the Regulatory Body after the relevant event occurrence. This report have to contain an event detailed description; a safety evaluation; direct and root causes; actions taken and, lessons learned;
- Also all unfavorable change to plant safety; weaknesses, degradation or impairment in the state or performance of the safety related system equipments and components must be reported;
- A quarterly report containing the non relevant events occurred during the considered period, according is established in the plant Operating Experience Management Program;
- A quarterly report with the results of the plant Operating Experience Management Program implementation;
- A quarterly report with the results of the plant Ageing Management Program implementation.

Nº 32

CNS-REF.-ART.: 9

PAGE OF REPORT: 45

CHAPTER OF NAT. REPORT: 3.9.2.

It is stated "In each NPP the Licensee shall appoint a person of this own body named Primary Responsible". How is it ensured that the primary responsible cannot be overruled in safety questions by headquarter management of the licensee's organization?

According to is established in the Mandatory Documentation (Operating Licenses and the Regulatory Standard AR 3.9.1.), the Responsible Entity (RE) is the licensee's organization responsible for the nuclear and radiological safety; the safeguards and the security of the plant. Besides, the RE has to monitor that the plant is operating in a safe condition and has to give the support to the plant manager needed for fulfil its responsibilities.

On the other side, the above mentioned Mandatory Documentation, establishes that the plants Primary Responsible (PR), who is designated by RE, is the direct responsible for the nuclear and radiological safety; safeguards and plant security, is the plant manager and can not delegate his responsibilities. In addition, the PR has to possess an individual license and a specific authorization (both regulatory permits are complementary and enable the PR to perform their specific functions). ARN routinely monitors the performance of the PR and the RE verifying, through regulatory inspections and audits, that it complies what is established in the mandatory documentation. Therefore, it is ensured that the PR cannot be overruled in safety questions by headquarter management of the licensee's organization.

Nº 33

CNS-REF.-ART.: 9

PAGE OF REPORT: 46

CHAPTER OF NAT. REPORT: 3.9.2.

The main idea of the Convention Article 9 is the requirement in setting by the state of principal responsibility for safety of a nuclear installation to a licensee holder and corresponding assurance of compliance with this requirement.

Is there a provision in the Argentinean legislation for primary responsibility of the licensee holder for the safety of the nuclear installation as is required by the Convention on nuclear safety and IAEA Standard No GS-R-1?

The Regulatory Standards AR 0.0.1., 10.1.1., details and prescribes the responsibility of the licensee holder as it is considered in article 9 and IAEA Standard GS-R-1. Moreover it is legally considered in the National Nuclear Law and in the obligations set in Vienna Convention on Civil Liability for Nuclear Damage.

Nº 34

CNS-REF.-ART.: 10

PAGE OF REPORT: 50

CHAPTER OF NAT. REPORT: 3.10.2.2.

Some countries use a method of poll of the personnel for an estimation of Safety Culture level. Do you apply such method?

Argentina does not use a method that includes polls. However, a systematic follow up allows qualitatively evaluating and also improving organization and personnel Safety Culture. ARN is in the way, according with international recommendations to apply in the near future methodologies that include polls.

Nº 35

CNS-REF.-ART.: 10

PAGE OF REPORT: 51

CHAPTER OF NAT. REPORT: 3.10.2.6.

With reference to section 3.10.2.6., Safety Culture Regulatory Assessment, could you please provide more information on any internal guidelines produced and / or any specific training provided to the regulatory staff performing inspections in order to assist them in recognising issues relevant to safety culture in the licensees' organizations, including signs of declining safety performance?

The regulatory staff responsible for the inspections receives training in Radiological and Nuclear Safety (courses sponsored by IAEA) which includes theoretical and practical safety culture issues, being used the corresponding IAEA guidelines.

Additionally, the staff receive an on the job training where experience is transmitted by senior regulators aimed to allow the staff to recognize the lack of safety culture. As part of the on the job training and to maintain a proactive attitude learning from the regulatory experience, when a possible indicator of safety culture lack is detected, it is promoted a meeting among the staff to discuss and analyze the issue and the proposal of the corresponding regulatory actions.

Nº36

CNS-REF.-ART.: 10

PAGE OF REPORT: 48 / 52

CHAPTER OF NAT. REPORT: 3.10.2.6.

The practical activities aimed at maintaining safety priority and safety culture are highlighted in the Report rather sufficiently. However, there is nothing said on establishment of the requirement to maintain the high level of safety culture in the legislative and regulatory framework. Please, provide your clarifications regarding this issue.

The safety culture principle is a fundamental principle and it is not prescriptive in legislation, because it lacks of a strict legal sense. However, in a lack of safety culture case, which imply a mandatory documentation non compliance, it will be taken into account in order to consider the grade of the sanction.

Nº 37

CNS-REF.-ART.: 11

PAGE OF REPORT: 53

CHAPTER OF NAT. REPORT: 3.11.1.

Argentina may please provide the details about the steps to ensure that adequate financial resources are available for the support of nuclear safety.

Nucleoeléctrica Argentina S.A. (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 Argentine Economy Ministry.

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

Apart from resources that come from electrical generation there are two national laws (Law Nº 24065 and Law Nº 24776) that guarantee all the needs for plant operation, maintenance and, the resources needed to support nuclear safety.

Nº 38

CNS-REF.-ART.: 11

PAGE OF REPORT: 56

CHAPTER OF NAT. REPORT: 3.11.2.3.

Can Argentina provide some details of measures taken to retain essential man power and knowledge management activities to preserve the corporate memory?

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

Nº 39

CNS-REF.-ART.: 12

PAGE OF REPORT: 62

CHAPTER OF NAT. REPORT: 3.12.3.

The report states that Argentina is taking into account human reliability relying on generic simulator (international simulator data). This approach will be very useful using a specific simulator.

Presently, the operation shifts training / retraining of the Argentine NPPs are not being carried out on its own operating plant simulators. As a consequence, the mentioned plant personnel training / retraining is being performed on simulators belonging to plants similar to CNA I and CNE making thus the time devoted to these activities, in some occasions, conditioned to issues related with logistic aspects and availability of the foreign simulators used.

Therefore, NA-SA has decided to improve the plant personnel training / retraining by providing CNA II and CNE with specific operating plant simulators. In view of that, it is foreseen to acquire both an interactive graphic simulator and a full scope simulator to be used for CNA II plant personnel training / retraining.

In a first stage, it is estimated that in the next two years an interactive graphic simulator will be available and in a second stage, a full scope simulator will be installed.

In the scope of CNE's life extension activities, aimed at incorporating the anticipated design modifications, a full scope simulator for plant personnel training / retraining will be provided and it is foreseen that it will be available in the near-future.

However, although as was mentioned the generic simulator used so far are not specific, according to the Argentinean plants experience, the following positive aspects that increase the reliability of human performance in a nuclear power plant can be mentioned:

- a. - Experience in the practice of acting in emergency situations, transients and plant malfunctions following the corresponding procedures;
- b. - Periodic training (in classroom) on the use of procedures, effective communication, prevention of human error techniques, plant operating experience; etc and;
- c. - Learn and see different work culture was very useful to improve the human performance in the plant operation.

Therefore, from the human reliability point of view, it could be concluded that would be beneficial to use a specific simulator.

Nº 40

CNS-REF.-ART.: 12

PAGE OF REPORT: 60 / 61

CHAPTER OF NAT. REPORT: 3.12.1.2.

It is stated “the periodic training of CNE personnel at Gentilly-2 simulator in Canada, CNA I personnel at Angra simulator in Brazil.” Where is the shift staff of Atucha 2 trained? Is there a plant specific simulator?

NA-SA has decided to provide Atucha II with a specific operating plant simulator. In view of that, it is foreseen to acquire both an interactive graphic simulator and a full scope simulator to be used for CNA II plant personnel training / retraining.

The supplier is TECNATOM. At the present time a delegation of engineers belonging to Atucha II staff is being specialized in Spain at TECNATOM.

In a first stage, it is estimated that in the next two years an interactive graphic simulator will be available and in a second stage, a full scope simulator will be installed.

Nº 41

CNS-REF.-ART.: 12

PAGE OF REPORT: 58 / 61

CHAPTER OF NAT. REPORT: 3.12.1.1. / . 3.12.1.2.

The initiatives taken by CNE and CNA management, for encouraging improvement in human performance through use of event free tools are commendable. Can Argentina share its experience in this area, specifically after new initiatives were taken?

The methodologies used are those employed at international level, therefore it is possible and convenient to share the experience in this area.

Due to the importance of benchmarking in the continuous improvement of human performance and the possibility of identifying items to be improved it is important for our plants to share experiences in the human development area. This has to be done according to common necessities and taking into account the available resources.

There is a continuous improvement program for human development.

For human performance improvement stress has been put on error reduction and task observation programs. Among the international techniques used for error reduction, the selection was made taking into account those that better fit plant necessities. In order to promote their use a periodic training was proposed at organization level and apart from that supervisors and area heads were encouraged to promote internal or external training of these techniques for relevant plant tasks.

With respect to task observation program the intention is to make area heads to go frequently to the field promoting in situ coaching of present tasks.

Nº 42

CNS-REF.-ART.: 12

PAGE OF REPORT: 58 / 60

CHAPTER OF NAT. REPORT: 3.12.1.1.

In section 3.12.1.1., Human Corrective Actions in CNE, information is provided on the implementation of the Human Performance Program. It is stated that "A better understanding of the results of the Human Performance Program can be learned through the information gathered from the management indicators." Could you please provide examples of performance indicators used to monitor the effectiveness of the implementation of the Human Performance Program?

Indicators that the Licensee uses to monitor effectiveness of the implementation of the Human Performance Program are the following:

- Industrial accidents rate/ Industrial accidents annual number;
- Distribution of individual dose and median individual dose / cumulative total collective dose;
- Report of the state of inadequate conditions (every four months);
- Internal events and related corrective actions (every four months);
- Activities related with technical information exchanges (every four months);
- Training hours programmed / performed training hours.

Other documents which are used as indicators are the periodic reports including conclusions on different processes such as internal operating experience, self assessment and task observations.

Nº 43

CNS-REF.-ART.: 12

PAGE OF REPORT: 58 / 62

CHAPTER OF NAT. REPORT: 3.12.1.1. / . 3.12.1.2.

In section 3.12.1.2., Human Corrective Actions in CNA, information is provided on the implementation of the Program for Human Performance Improving. It is stated that "Prior to the scoping and contents definition of the Program for Human Performance Improving, benchmarks were established against the Spanish Vandellós and Ascó NPPs, and the Mexican Laguna Verde NPP." Provided that a similar program is implemented in CNE, are there any planned benchmarking activities between these two operating organizations to compare practices and share experience on the implementation of the human performance improvement programs?

Both plants CNA I and CNE belong to the same operating enterprise Nucleoeléctrica Argentina S.A. (NA-SA) and for that reason they use similar procedures and there is very good exchange experience among plant personnel as well as with the Plant Services Support sector of the enterprise.

Benchmark activities are periodically performed between the two NPPs at different levels of the organizations and are included among the corresponding annual objectives of each NPP. They are coordinated by the deputy general manager of the productive area of the enterprise.

This activity is fulfilled performing technical exchange with other plants and institutions, through seminars, training courses and technical meetings sponsored by IAEA, WANO, COG and others. In addition, plant personnel participate in WANO peer reviews and technical support missions on different subjects.

Nº 44

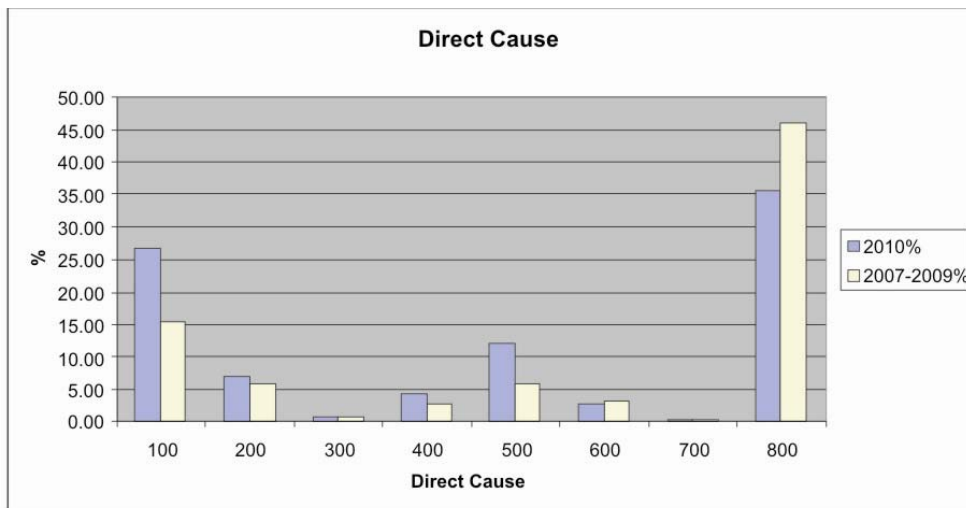
CNS-REF.-ART.: 12

PAGE OF REPORT: 60

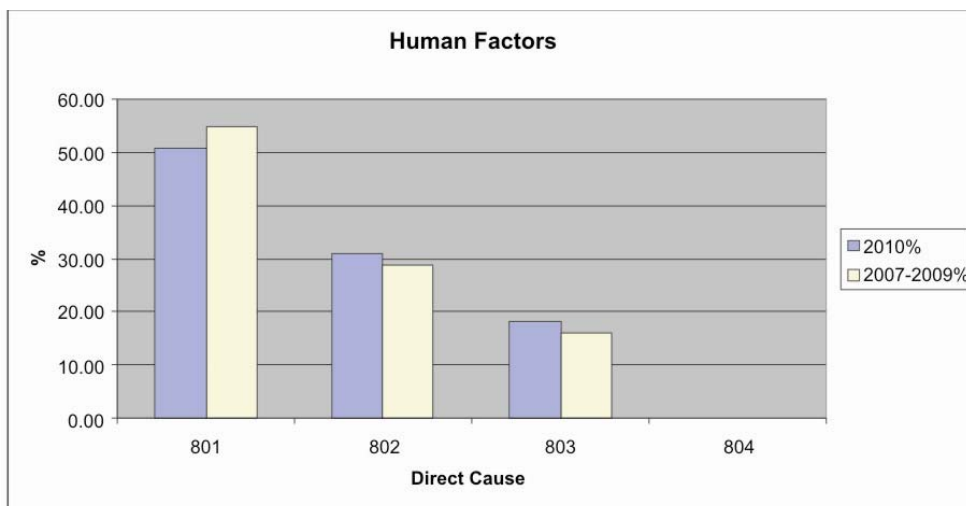
CHAPTER OF NAT. REPORT: 3.12.1.1. / . 3.12.1.2.

In August 2008 Atucha I NPP started to implement a Program for Human Performance Improving. The report does not contain statistical data on events (accidents/incidents) at NPPs and the ratio of human induced events (incidents). Could Argentina provide these statistical data for the reporting period?

The statistical data requested for Atucha I NPP are the following:



Direct cause: Direct cause event distribution according to WANO classification



Human Factors according to WANO classification

WANO designation

Code	Description
100	MECHANICAL DEFICIENCY
200	ELECTRICAL DEFICIENCY
300	CHEMICAL or CORE PHYSICS DEFICIENCY
400	HYDRAULIC AND PNEUMATIC DEFICIENCY
500	CONTROL AND INSTRUMENTATION DEFICIENCY
600	ENVIRONMENTAL (ABNORMAL CONDITIONS INSIDE THE PLANT)
700	ENVIRONMENTAL (EXTERNAL TO THE PLANT)
800	HUMAN FACTORS
801	Slip or lapse
802	Mistake
803	Violation
804	Sabotage

Nº 45

CNS-REF.-ART.: 12

PAGE OF REPORT: 58 / 63

CHAPTER OF NAT. REPORT: 3.12.1. / . 3.12.4.

"The report mainly deals with the QA organization of the licensees. Could Argentina present details about the audits carry out at the suppliers/manufacturers?"

The supplier control of the NASA QA System is based on written procedures. NA-SA evaluates the suppliers to verify their aptitude in manufacturing components / equipments or bringing a service. Furthermore NA-SA participates in inspections and tests during the components / equipment manufacturing to verify the corresponding Technical Specifications fulfilment.

These activities are carried out by QA personnel and experts in the specific areas involved. The scope of such activities will depend on the item safety relevance.

However, it should be noted that the suppliers of safety related equipment, components and services are the designers themselves, keeping consequently the original technical specifications as well as the corresponding quality requirements.

Nº 46

CNS-REF.-ART.: 13

PAGE OF REPORT: 65

CHAPTER OF NAT. REPORT: 3.13.2.

It is stated in the report that QA Programs for NA-SA, CNA I and CNE fulfill the requirements of Regulatory Standard AR 3.6.1., IAEA Practice Code 50-C-Q and all IAEA applicable Safety Guides. How is the process-based approach implemented according to the international requirements and recommendations (IAEA GS-R-3, GS-G-3.1, GS-G-3.5, ISO 9001/ 14001)? Is it the regulatory requirement that the licensees shall have their QMS certified for compliance with ISO 9001/14001 and OHSAS 18001?

The NA-SA Quality Management System (QMS) is based on the IAEA Code of Practice 50-CQ and its corresponding Safety Guidelines. In relation with the process-based approach, it is foreseen to update such QMS according to the IAEA GS-R-3 recommendations.

Besides, to fulfill a requirement of the Electrical Power Regulatory Body, NA-SA has implemented and certified in 2006 its Environmental Management System for compliance with ISO 14001.

The certification for compliance with ISO 9001/14001 and OHSAS 18001 is not a regulatory requirement. However, the regulatory standard AR 3.6.1. requires that the Licensee must establish an adequate quality system compatible with the organization's Quality Policy which includes safety, environmental and health aspects.

Nº 47

CNS-REF.-ART.: 13

PAGE OF REPORT: 64

CHAPTER OF NAT. REPORT: 3.13.2.

See also 3.19.6. Engineering and technical Support page 122. The report states (p65) that "The Licensee verifies that every person and every organization involved in the nuclear area becomes thoroughly familiar with QA requirements." This and section 3.19.6. recognises that contractor's (and people or organizations external to the licensee) activities can have an influence on safety. I) How does the licensee ensure that it has the capability to understand advice and service given to it and the context, for safety, in which that advice sits? II) How does ARN ensure that the licensee has, and takes steps to retain, adequate capability within its own organization to understand the nuclear safety requirements of all of its activities relevant to safety, and those of contractors and not delegate to contractors responsibilities which are properly those of the licensee?

I) An adequate capacity of NA-SA staff to understand the nuclear safety requirements, furthermore to interpret the safety philosophy, is accomplished through academic education, field training, exchange of experience with plant operators that are referenced for their excellent performance and maintaining fluid relations with recognized international organizations.

In this sense, NA-SA is concerned that its personnel perform the necessary courses to strengthen their skills, participate in international courses and workshops, to train entrants through the daily support in their work areas by experienced personnel, to encourage participation in peer reviews in

nuclear power plants around the world through WANO and other organizations such as the CANDU Owners Group.

Moreover, NA-SA has cooperation agreements with several local and foreign institutions, allowing its staff to be in touch with international experts. The implementation of these measures avoids the isolation of the operators, promotes the use of good practices and ultimately allows the development of a culture of operation according to international reference operators.

Besides the above-mentioned and to promote conservative decision making collectively, NA-SA has established internal organizations in both plants (PIAC, Plant Internal Advisory Committee) and in the headquarter (TRC, Technical Review Committee), to analyze all issues that may affect nuclear safety. Plant specialists in each area related to safety make up the PIAC in each plant advising the plant manager, while specialists belonging to headquarter together with the plant managers make up the TRC advising the enterprise general manager.

Modifications to the facility that affect the original design, events occurred during operation, planning of tasks involving abnormal risks and periodic assessment of each plant area are issues to be analyzed in the above-mentioned committees.

II) The ARN carry out different, inspections (routine and special); audits and regulatory assessments with the purpose of verifying that nuclear installations satisfy the standards, licenses and requirements in force.

Concerning the QA requirements related with the persons and organizations which activities can have an influence on safety –including contractors / organizations external to the licensee–, the ARN through inspections and audits, which implies to witness in a close follow up the training and qualification of the personnel and the remaining activities relevant to safety, verifies that the QA Program of the licensee is adequately implemented and is aimed to achieve and retain within its organization the necessary capability to understand the nuclear safety requirements of the activities relevant to safety, including those contractor's activities.

Besides, the ARN verifies that the licensee implements an inspection / audit program that covers the contractors' activities / external organizations, assuring that licensee responsibilities are not delegated.

Nº 48

CNS-REF.-ART.: 14

PAGE OF REPORT: 83

CHAPTER OF NAT. REPORT: 3.14.1.

The reports states that “the indicators evaluation was made throughout the analysis of different parameters”. Could Argentina give more details about the use of a precursor analysis program as a safety indicator (probabilistic analysis of events)?

ARN does not consider the use of precursors in the performance indicators programme. However, in NA-SA and ARN there are groups dedicated to analyze the operative experience in national and foreign NPPs. When it is applicable to Argentine NPPs the information is sent to the groups dedicated to PSA for the precursor analyses.

Nº 49

CNS-REF.-ART.: 14

PAGE OF REPORT: 82

CHAPTER OF NAT. REPORT: 3.14.3.2.

Could Argentina provide more details concerning PSA use and development: are there Probabilistic Safety Goals? Do the recently updated PSAs take into account new problems like aging, new technologies (digital I&C), climate change? Could Argentina give more details on the precursor analyses?

PSA have been developed with sufficient detail in order to be used for applications.

The probabilistic acceptance criterion for NPP's design is established in the regulatory standard AR 3.1.3. A level 3 PSA must be developed to fulfil this criterion, which focuses attention on the individual risks due to individual doses estimated for each postulated accident sequence.

Probabilistic risk assessment is included in the evaluation of all design changes whenever appropriate, in the comparison of design alternatives, prioritization of components in maintenance programmes, definition or review of test intervals and allowed components or systems outage times. Both plants have made a probabilistic assessment in order to extend the outages periods from 12 to 18 months.

Although the regulatory standards are probabilistic in nature, there is not a probabilistic goal established for CDF. However, for direct applications of level 1 PSA results, international reference values are considered. For instance, up to 10^{-4} / y is considered an acceptable value for the total CDF

for NPPs in operation and a maximum variation of 10^{-5} for the CDF due to change in operating conditions / design modifications.

Atucha I PSA has been updated in order to reflect the actual condition of the plant. Level 1 PSA is in its forth revision, design changes have been incorporated and new technologies have been taken into account as new digital I&C. Operating experience is always updated, in particular all plant outages were analyzed up to the year 2007 completed for revision 3. New operating procedures, some of them recently developed within the scope of Accident Management program, were also incorporated.

Embalse PSA Level 1 has a detailed model which will be updated in order to include all modifications related to the PLEX project. As other CANDU plants Embalse has probabilistic reliability goals for safety systems. PSA fault trees are used to calculate the actual reliability taking into account plant data.

The PSA of Atucha I and Embalse NPPs do not include any effect about climate changes up to now. However, some of the possible effects could be included in operating experience, affecting the frequency of some initiating events in future PSA upgrades.

With regard to the inclusion of new issues in PSA upgrades, we must say that aging is not taken into account directly in the model. This effect is considered in the upgraded PSA versions through the use of the new registered operating experience.

PSA groups has a close relation with the operating experience groups and uses the reports of the events in order to update initiating events and its frequencies as well as data reliability, in particular operator actions. Precursor analysis program begun formally in 2010 and the data corresponding to 2008 / 2009 are being analyzed.

Nº 50

CNS-REF.-ART.: 14

PAGE OF REPORT: 81

CHAPTER OF NAT. REPORT: 3.14.3.2.

If a Probabilistic Safety Assessment (PSA) is performed, will low power operation be part of it?

The low power PSA's were developed separately in all cases. This means they are not included in a unified model. Updates are also separate.

In case of the PSA applications mentioned on page 82 of the National Report reference document, the full power PSA model was used. These models are developed for internal events and full power condition only. However, when decisions must be based on acceptance criteria in some applications, conservative safety margins are included to consider the missing contributions.

Nº 51

CNS-REF.-ART.: 14

PAGE OF REPORT: 75

CHAPTER OF NAT. REPORT: 3.14.3.1.3.

Argentina report says "The first phase of the PLEX program has to identify the necessary modifications and updates of the plant, in order to ensure a safe operation for an extended period of 25 – 30 years." How did NA-SA verify the extended period achievable? Did NA-SA set up the extended period of 25 – 30 years by feasibility study which ensured the safety operation prior to the first phase?

The life extension period of 25 to 30 years is considered feasible because the critical reactor components that limit the life of a CANDU plant are the fuel channels and the steam generators, and both of these components will be replaced as part of Embalse NPP refurbishment. So it is assumed that the plant would be in position to operate for an additional period similar to the original design life. This assumption is endorsed by the plant designer, AECL.

Besides, regarding other plant structures systems and components, the corresponding prognosis for end of life and long term operation were obtained from life and condition assessment studies.

Nº 52

CNS-REF.-ART.: 14

PAGE OF REPORT: 75

CHAPTER OF NAT. REPORT: 3.14.3.1.3.

Does a CNE life extension feasibility study includes spent nuclear fuel management aspects and is there any strategies prepared regarding spent fuel management, in case if the CNE unit lifetime will be extended? Does the Regulatory Body of Argentina requires Preliminary Decommissioning Plan at the very beginning of the Nuclear Power Plant lifecycle, as there is

no information presented in National Nuclear Safety Report 2010 regarding decommissioning aspects of the CNA I, CNE or the new CNA II unit, which is under construction?

The Embalse spent fuel dry storage system (ASECQ) design lifetime is 50 years and considering that the ASECQ system began to operate in 1993, it means that would reach its end of life by 2043. At present the system installed capability is of 216 silos and it has an enough storage capability to cover the spent fuel storage requirements of both the remaining plant life and the future extended life which would reach its end of life before 2043.

A Preliminary Decommissioning Plan is part of the Mandatory Documentation which is included in the Operation Licenses of CNE and CNA I NPPs. For CNA II NPP, decommissioning aspects are included in the corresponding Final Safety Report.

Nº 53

CNS-REF.-ART.: 14

PAGE OF REPORT: 73

CHAPTER OF NAT. REPORT: 3.14.3.1.2.

The report states that “passive autocatalytic hydrogen recombiners “will be installed at CNA II. Does the operator consider to install in addition hydrogen igniters? If yes at which concentration the igniters will be activated? Which computer code was used by GRS to calculate hydrogen release and recombination capability?

Only passive autocatalytic hydrogen recombiners will be installed at CNA II. No additional hydrogen igniters are foreseen.

The hydrogen release and distribution inside the containment was calculated by GRS with detailed nodalizations of the MELCOR code. Later together with AREVA, taking into account the similarity between CONVOI plants and CNA II NPP, the distribution (position and capacity) of the recombiners was decided.

The recombiners capability is for the calculation performed by GRS an input which was provided by the manufacturer AREVA.

Nº 54

CNS-REF.-ART.: 14

PAGE OF REPORT: 81

CHAPTER OF NAT. REPORT: 3.14.3.2.

Could you please specify the average number of man-days necessary for PSA review by regulatory authority? Is PSA review done by regulatory authority only or any external resources/experts are also applied?

Considering as example our experience during the assessment of PSA level 1 of CNA II, currently under development, we have used the “on-line” methodology. This has involved a large investment of time, since many reports were then re-evaluated, but implies less effort required to review the final report. To review the final report, we estimated a total of 300 man-days. This estimation includes use of external support provided by experts of different topics.

Nº 55

CNS-REF.-ART.: 14

PAGE OF REPORT: 83

CHAPTER OF NAT. REPORT: 3.14.4.

The report states “It should be mentioned that Argentine experience in the use of Safety Performance Indicators 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 safety performance indicators is used as a regulatory tool to provide an additional view of the NPPs performance, allowing to improve 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. ”UK strongly supports the final sentence. Argentina is requested to provide any safety performance indicators used by ARN and the licensees. Could ARN comment on how it uses the SPIs in making judgments about licensees’ safety?

The regulatory plant safety performance indicators used are the following:

- PLANT STABILITY:

1) Forced outages:

$$\frac{\text{Number of unplanned shutdowns} \times 100}{\text{Load Factor}}$$

2) Forced power reductions:

$$\frac{\text{Number of unplanned power reductions below 95\% full power} \times 100}{\text{Load Factor}}$$

3) Load Factor:

$$\frac{\text{Energy generated}}{\text{Maximum possible generation}} \times 100$$

- RADIOLOGICAL PROTECTION:

4) Individual maximum dose: Maximum monthly dose in the period (mSv)

5) Total equivalent dose: Total equivalent dose in the period (Sv – M)

6) Liquid effluent discharges: Percentage of the quarterly discharge limit

7) Gaseous effluent discharges: Percentage of the quarterly discharge limit

8) Radioactive wastes: Volume [m³] of low activity solid wastes generated in the period.

- SURVEILLANCE:

- Maintenance:

9) Corrective work orders issued: Number of reports on safety or safety-related system deficiencies of a corrective nature submitted during the period

10) Pending corrective tasks: Number of reports on safety or safety-related system deficiencies of a corrective nature which are still pending, excluding those requiring the cool shutdown of the nuclear power plant

11) Pending corrective tasks because of lacks of supply: Number of reports on safety or safety-related system deficiencies of a corrective nature, which are still pending due to lack of supplies

12) Reworking: Number of reports on corrective-type deficiencies in the safety or safety-related system components subject to corrective or preventive maintenance in the previous six months

13) Overdue preventive maintenance tasks: Number of overdue preventive or predictive routine inspections and maintenance tasks involving safety or safety-related system components, excluding those requiring the cool shutdown of the power plant.

- Repetitive Tests:

14) Overdue repetitive tests: Number of overdue repetitive tests of safety or safety-related systems

15) Failure discovered by repetitive tests: Number of deficiency reports submitted on the basis of the repetitive tests performed on safety or safety-related systems

16) Test procedures under revision: Number of test procedures whose revision or issuance is overdue

- ORGANIZATION:

- Training.

17) Training: Number of hours devoted to training on safety-related issues

Operating Experience

18) Operating experience: Number of documented event analyses, findings or design modifications in similar power plants

- Internal control

19) Internal control: Number of internal technical audits

- Compliance

20) Pending Regulatory Requirements: Number of pending Regulatory Requirements

- ABNORMAL OPERATION

21) Significant events: Number of significant events

22) Safety Systems actuations: Number of automatics actuations of Safety Systems

- RISK

23) Safety Systems unavailability

24) Impact of reported events on core damage frequency.

As was mentioned in our National Report the SPIs contribute to have an estimation of the plants safety status and tendencies.

In case of possible early safety degradation indication coming from the tendencies or values of the SPIs, as the corresponding regulatory action which will vary depending on the kind of safety degradation, it is foreseen to carry out inspections, audits and/or assessments.

As a result of the above mentioned regulatory action, aimed to confirm the safety degradation existence, it is expected to know details about the degradation, direct and root causes, to estimate its evolution and the action taken or foresees to be taken in future by the Licensee.

Finally, based on the knowledge available about such safety degradation, the Regulatory Body will decide the necessary requirements to the Licensee correct its degradation and avoid its recurrence.

Nº 56

CNS-REF.-ART.: 14

PAGE OF REPORT: 80

CHAPTER OF NAT. REPORT: 3.14.3.1.6.

How many SG tubes have leaked so far at CNE? What is the design margin in number of tubes?

- a) In all four SG we had a total of 13 leaks since start of operation, most of them due to fretting at the U-bend. After implementing all corrective actions we have operated two complete operating periods without any leak. Preventively, we have plugged 2.7% average of total of tubes in each SG.
- b) By design, at zero fouling factor the margin is 25% of original heat transfer area (about 888 tubes). At design fouling factor of 0.0002 hr ft² °F/BTU, the margin should be 5,6% (about 200 tubes).

An engineering analysis was performed to determine the margin as per 2005 condition. As a result a theoretical margin of 18% (640 tubes) was obtained as of 2005, but a practical limit of 10% (355 tubes) was adopted to implement some corrective actions.

Nº 57

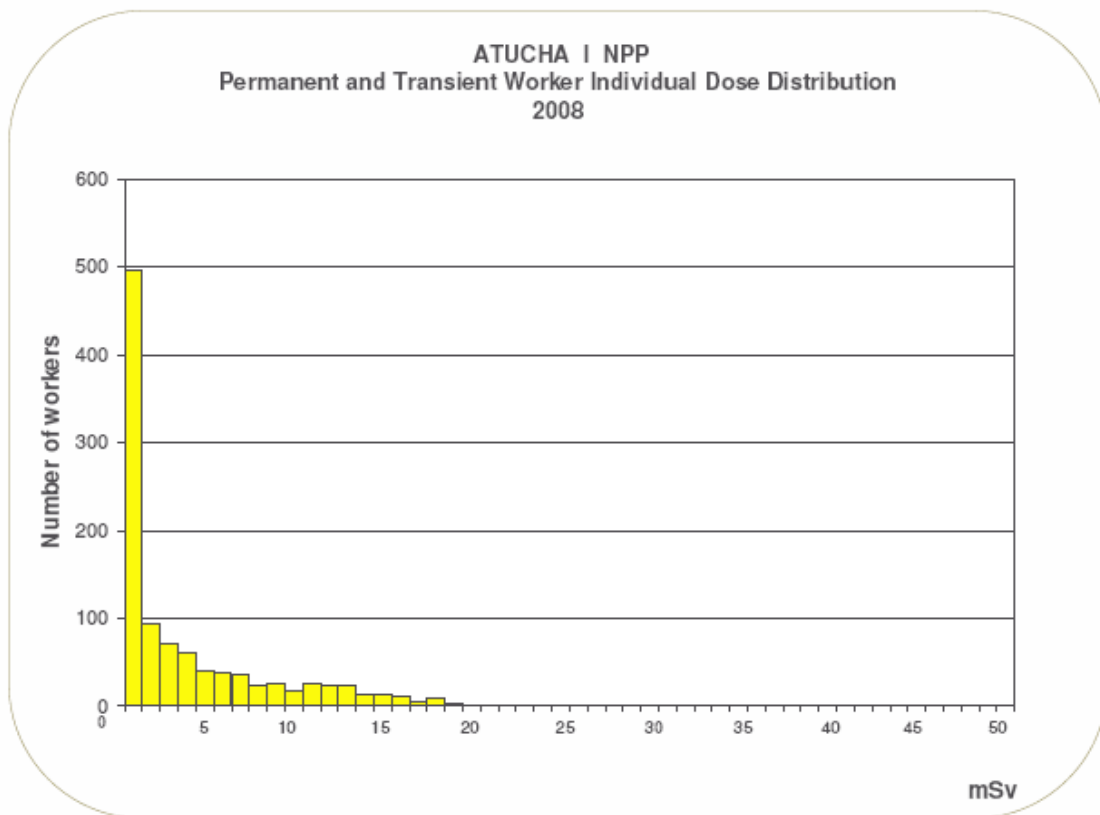
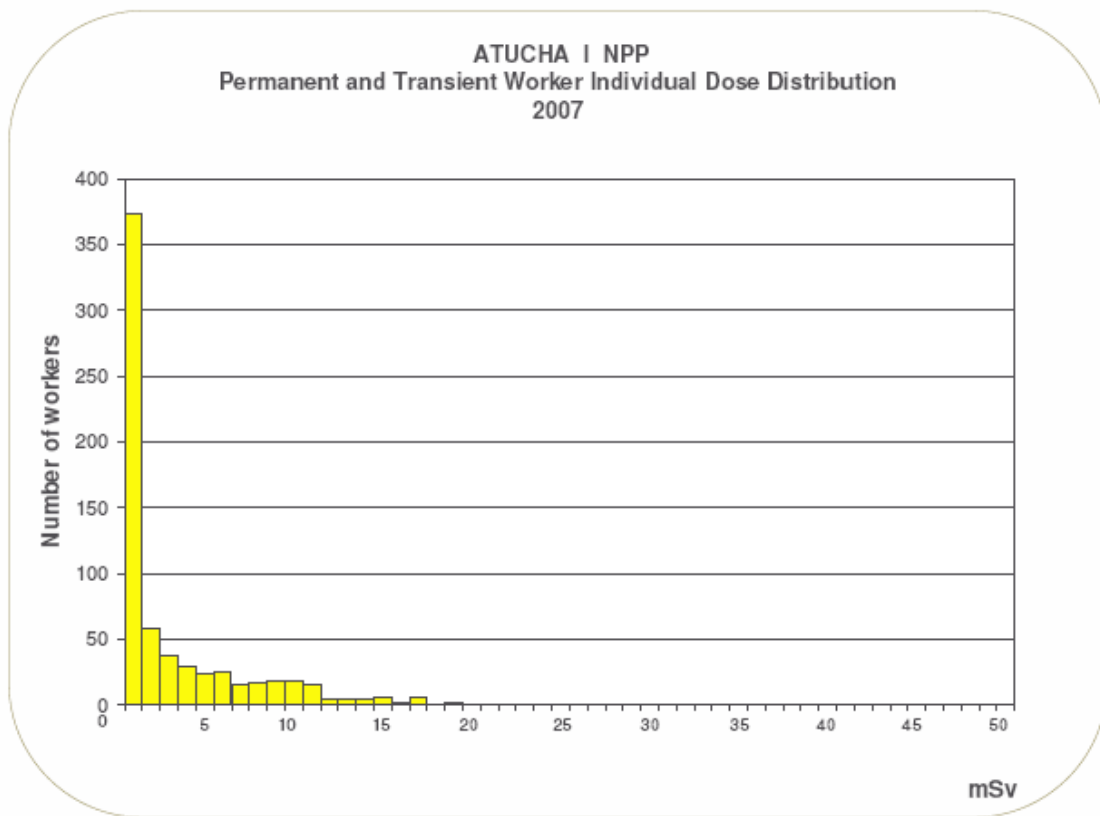
CNS-REF.-ART.: 15

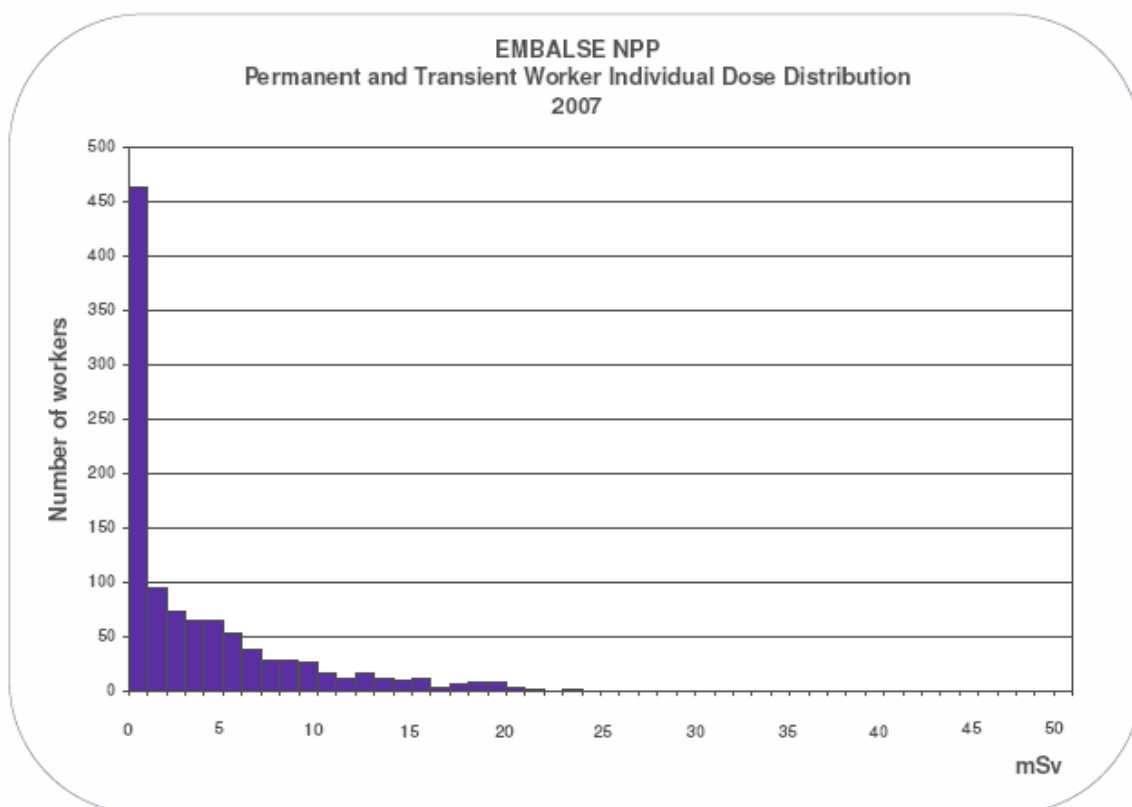
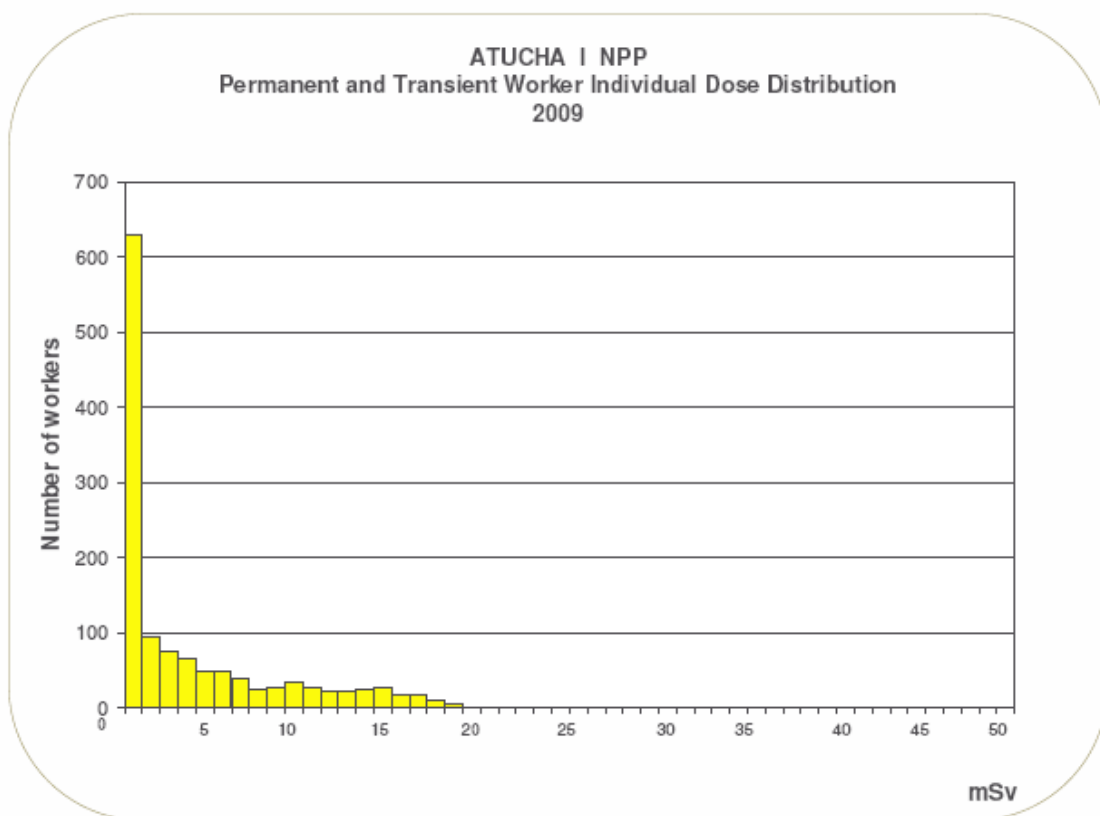
PAGE OF REPORT: 87

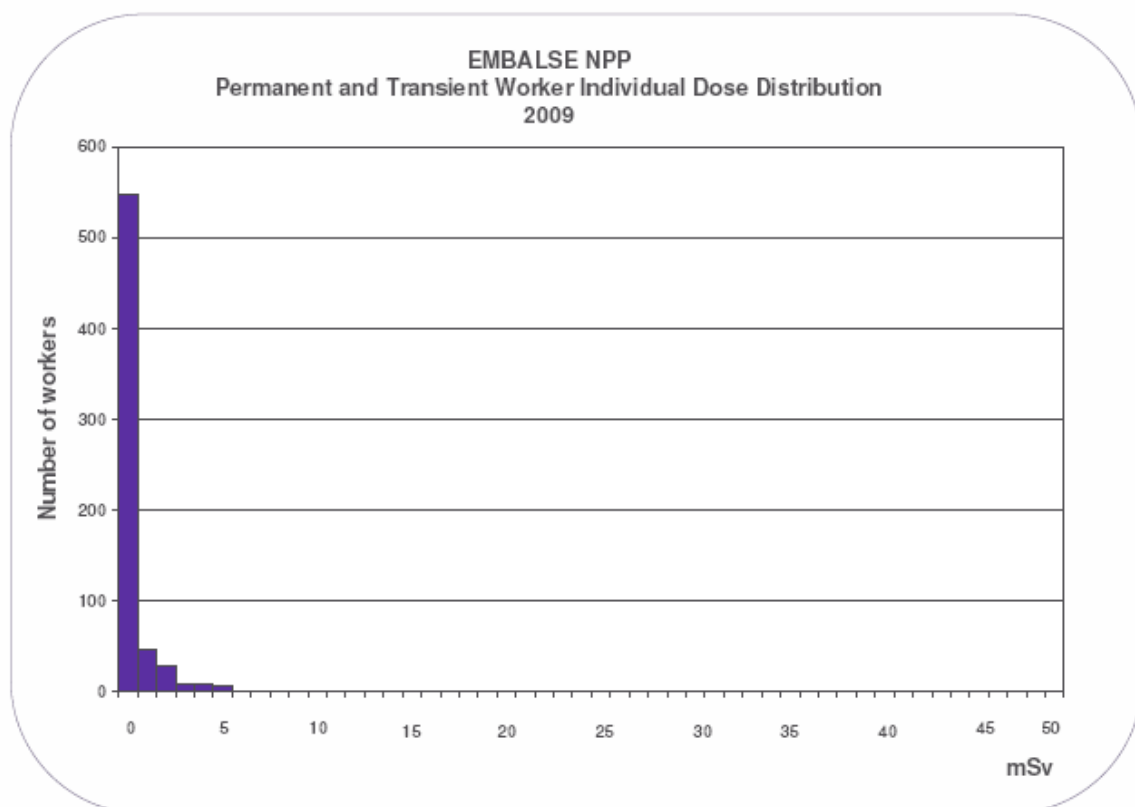
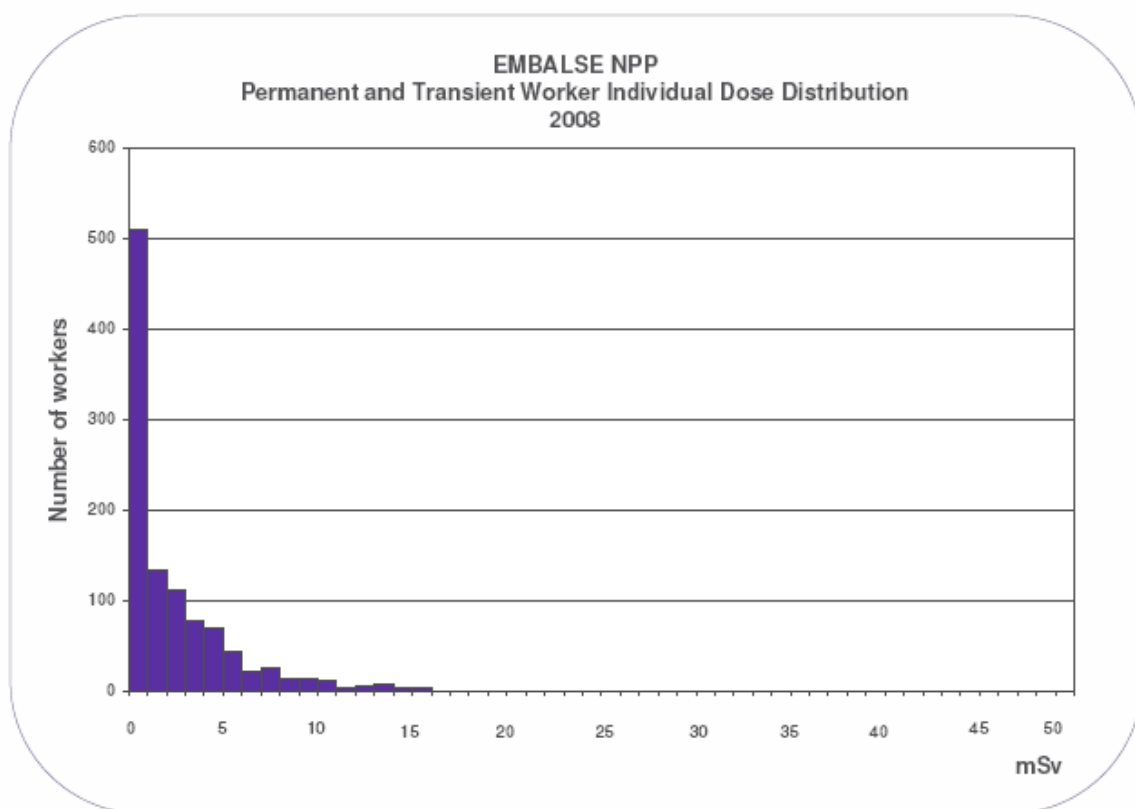
CHAPTER OF NAT. REPORT: 3.15.4.

Could Argentina present a diagram indicating the individual dose distribution for workers per year? Could Argentina give some elements about the intern effective dose for occupational workers?

- a) The following graphics show the annual individual dose distributions of the permanent and transient workers for each NPP.







- b) For Atucha I, the contribution of the intern effective collective dose to the total effective collective dose is 16% as annual average. For Embalse it is 35% as annual average. Almost all internal doses are due to tritium incorporation.

N° 58
CNS-REF.-ART.: 15
PAGE OF REPORT: 84
CHAPTER OF NAT. REPORT: 3.15.2.1.

What is the definition of the “critical group”?

Critical group is the population group representative of the most exposed individuals which are homogeneous regarding the parameters that influence on the received doses during normal operation of a given facility.

N° 59
CNS-REF.-ART.: 15
PAGE OF REPORT: 85
CHAPTER OF NAT. REPORT: 3.15.2.2.

Please specify “very conservative hypothesis”.

Very conservative hypothesis means to consider that the whole foodstuff consumption of the most exposed individual are locally produced.

N° 60
CNS-REF.-ART.: 15
PAGE OF REPORT: 87
CHAPTER OF NAT. REPORT: 3.15.4.

Are dose limits defined for the occupational exposure of trainees, students and pregnant women?

For students of age 16 to 18 who are required to use sources in the course of their studies, the annual limit of effective dose is 6 mSv in a year. It's not admitted occupational exposure for people younger than 18 years old.

There are not defined occupational dose limits for trainees.

For pregnant women, with the objective of not exceeding in the fetus the public dose limit there must be established worker conditions in order to try to avoid receiving a personal equivalent dose on the abdomen surface exceeding 2 mSv, and that every radionuclide incorporation will not exceed 1/20 of the respective annual incorporation limit, during all the remaining pregnant period.

N° 61
CNS-REF.-ART.: 15
PAGE OF REPORT: 88
CHAPTER OF NAT. REPORT: 3.15.5.

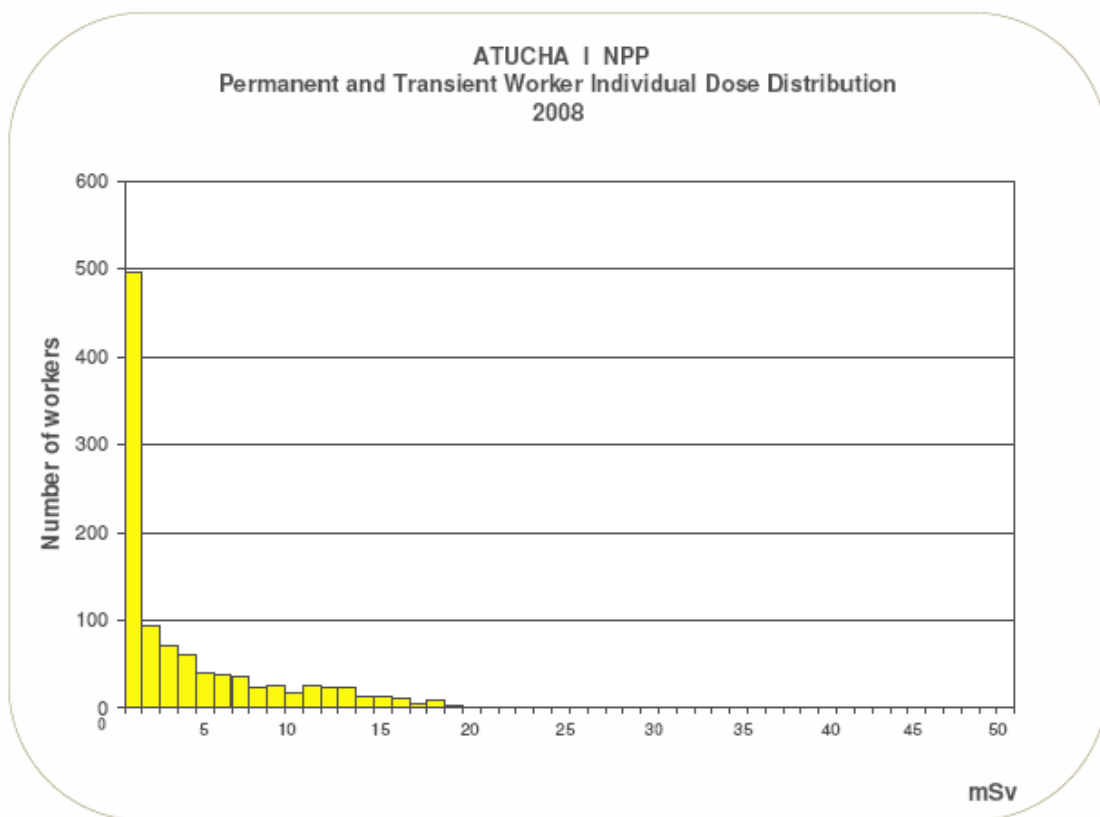
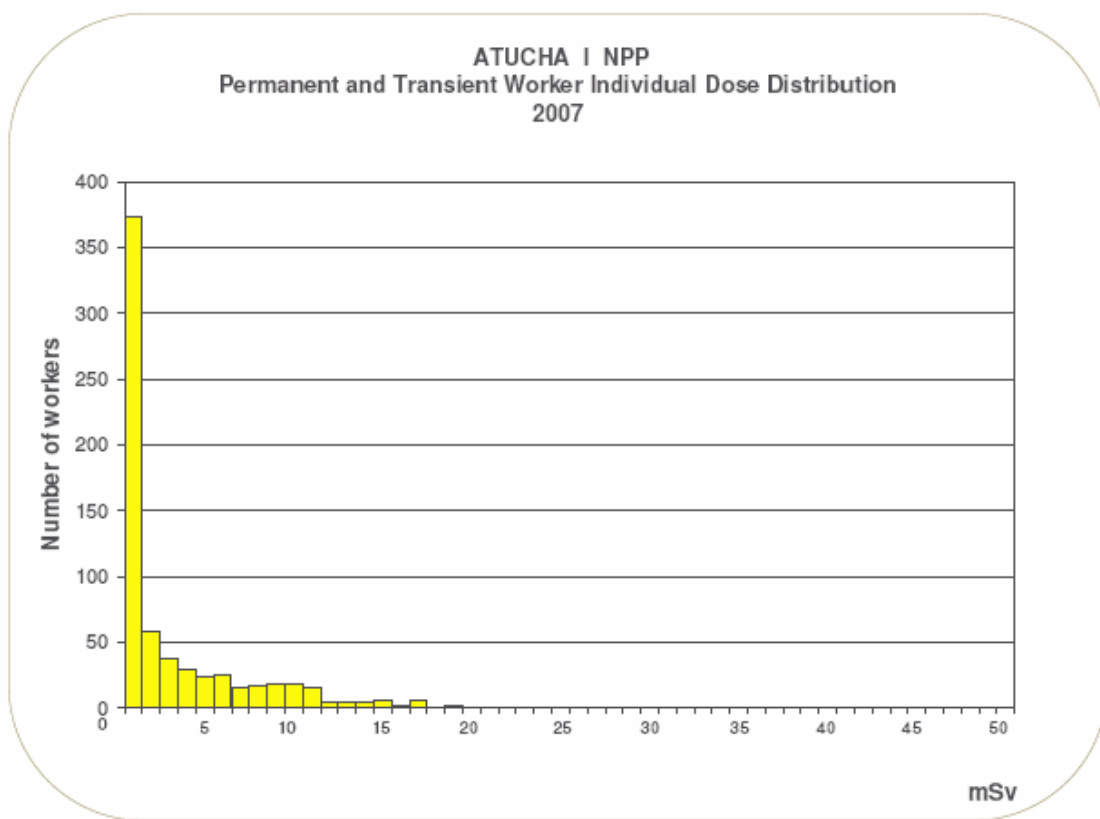
Please specify “significant doses”, for which detailed planning of activities is performed.

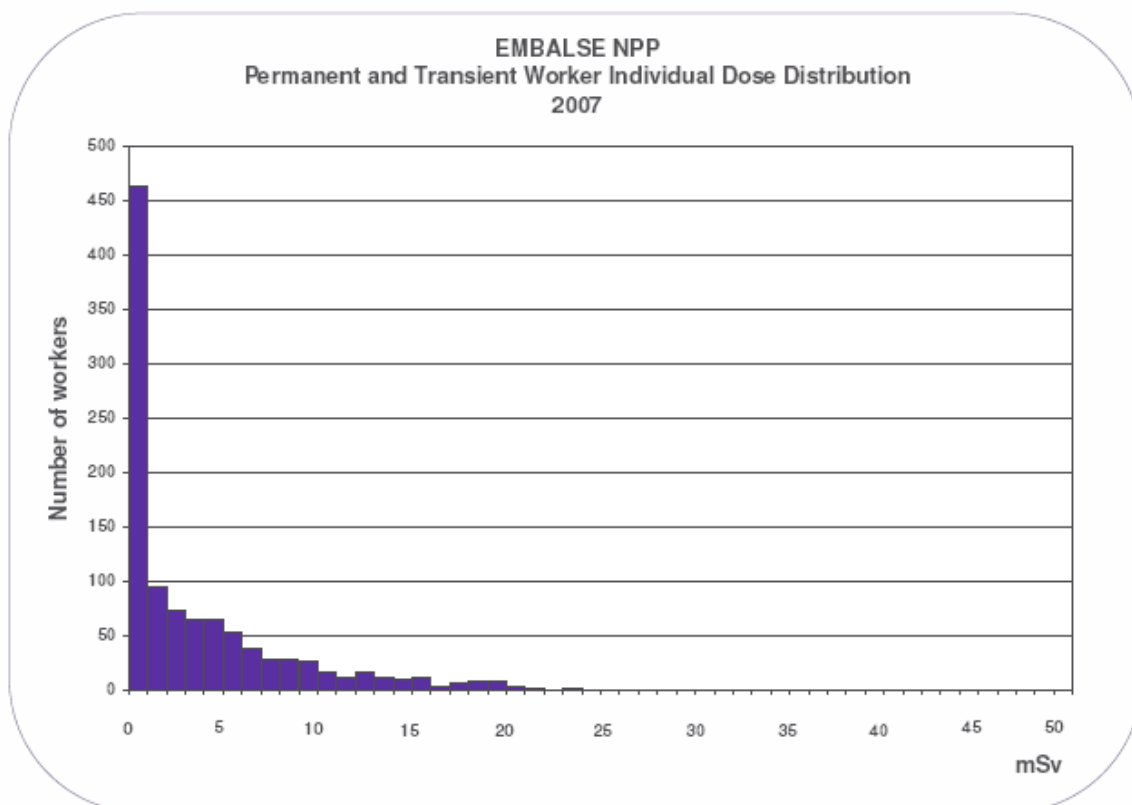
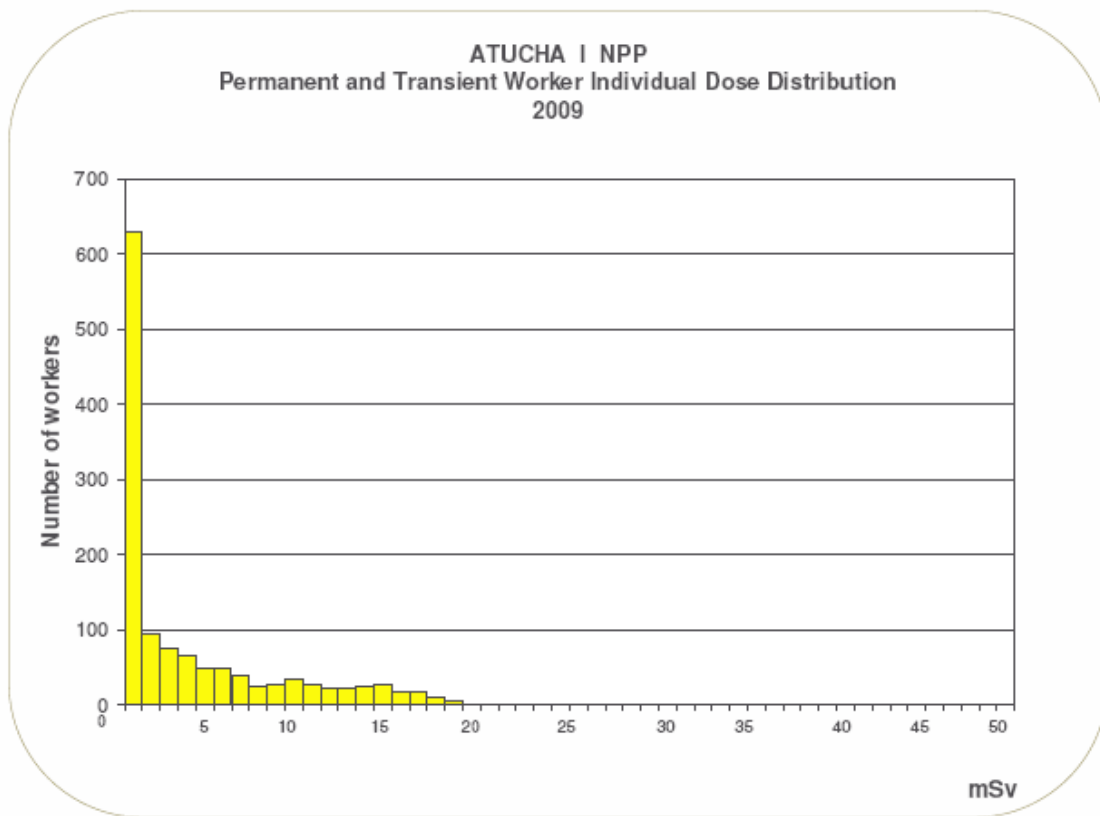
Detailed planning of activities is performed when it is expected that the exposure rate in the place is over 5 mSv / h or the expected individual dose is over 5 mSv.

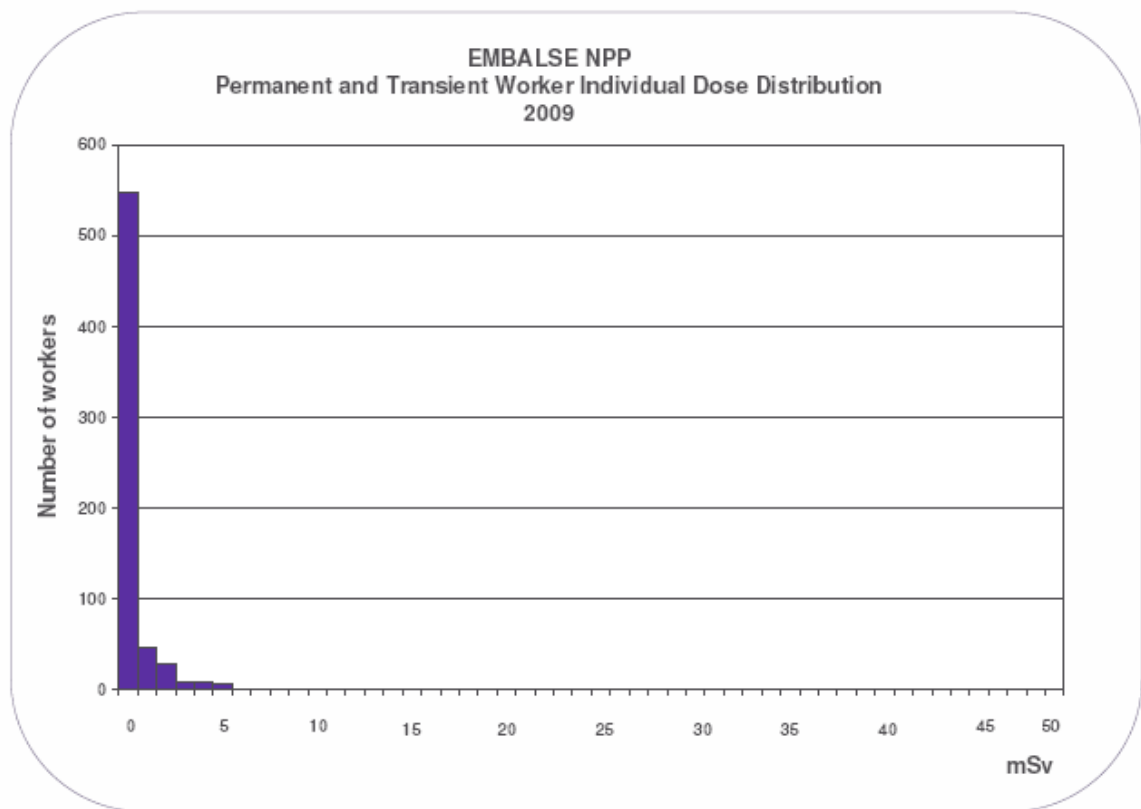
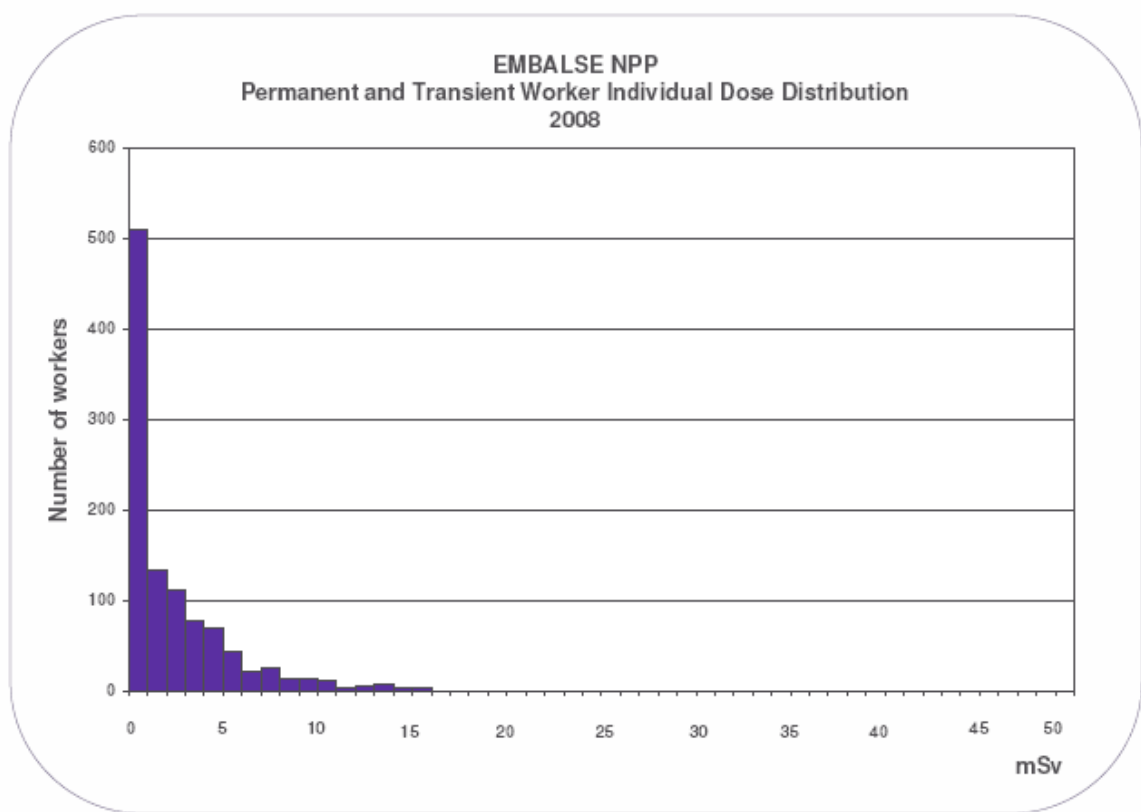
N° 62
CNS-REF.-ART.: 15
PAGE OF REPORT: 87 / 88
CHAPTER OF NAT. REPORT: 3.15.4.1.1.

What is a range of annual individual dose values recorded for the overwhelming majority of NPP workers?

The annual individual dose distributions of the permanent and transient workers for each NPP are showed in the following graphics:







Nº 63
CNS-REF.-ART.: 15
PAGE OF REPORT: 87 / 88
CHAPTER OF NAT. REPORT: 3.15.4.1.1.

Do the data on exposure (provided in the Report) include the internal doses? If that is the case, what is its percentage?

Yes, exposure data include internal doses. For Atucha I, the contribution of the intern effective collective dose to the total effective collective dose is 16% as annual average. For Embalse it is 35% as annual average. Almost all internal doses are due to tritium incorporation.

Nº 64
CNS-REF.-ART.: 16
PAGE OF REPORT: 97
CHAPTER OF NAT. REPORT: 3.16.12.

Nuclear emergency drills are regularly carried out with the participation of the population. Could Argentina give more details about participation of the population? Could Argentina present lesson learnt of this action?

The population surrounding the NPP voluntarily participates in each drill using protective actions to take in case of nuclear emergency. These are: early evacuation, access control, sheltering and iodine prophylaxis.

People and students participate in preparatory meetings, where the responsible for the drill pick up the comments and discussions about the exercise development.

Local radios are one of the main media used to inform about the exercise development, and the population uses this media to express their concerns. Many of these concerns have helped to improve the implementation of protective actions, such as the distribution of iodine tablets, the distribution of brochures during the access control.

Nº 65
CNS-REF.-ART.: 16
PAGE OF REPORT: 91
CHAPTER OF NAT. REPORT: 3.16.2.

It is stated that the operational capacity of the ARN in the management of the implementation of actions to protect the members of the public in the surroundings of the NPPs from radiological consequences is currently checked by carrying out the emergency exercises. Since this was already stated in the last report, are there new findings concerning this topic?

Based on the findings of recent exercises conducted in each NPP, some weaknesses were found. In this sense, there was a need to standardize the equipment used for the determination of concentrations in air and currently we are working on it.

On the other hand, another need identified was the preparation of communications with the media in an emergency. As a result, the First Workshop on Public Communication in Nuclear Emergencies in September 2010 was held and the staff in areas of communication and nuclear emergencies of NPP attended.

Nº 66
CNS-REF.-ART.: 16
PAGE OF REPORT: 93
CHAPTER OF NAT. REPORT: 3.16.6.

Can Argentina specify the requirement set by ARN for periodic review and update of emergency plans?

Important contributions to a continuous improvement of the emergency plans are the lessons learnt from each emergency exercise which is carried out with a two years frequency. After the exercise a meeting between personnel from the plant and the Regulatory Body is held to discuss and identify exercise weaknesses as to incorporate the lessons learnt updating the emergency plan if the regulator considers it as necessary. Besides, the plant emergency plans are reviewed and updated once the operating licenses expire.

The Report allows making a conclusion that the operating organization in Argentina is responsible for the activities not only on the NPP site but also beyond it. Surely, this is the issue of national jurisdiction and experience; moreover, the IAEA recommendations allow various options. However, it seems to be unreasonable to impose on NPPs some functions, which is regularly not attributable to them; this, on the one hand, decreases NPP capabilities to resolve its main tasks, including protective actions inside the site, and, on the other hand, decreases the responsibilities of the national authorities at all levels in the area of protection of the population and environment. Please clarify your position with respect to this issue.

In a nuclear emergency situation in Argentina, the Primary Responsible of the NPP (plant manager) shall declare the emergency and is responsible for the conformation and operation of the Local Emergency Operative Center (COEM) until the ARN arrives to the place and takes the control of the response. This is stated in the existing plans of NPP approved by the ARN.

As follows from the Report, the Argentinean Regulatory body ARN is responsible for approval of program of actions in case of nuclear accidents, emergency preparedness programs, programs of emergency preparedness training for personnel and general public. Moreover, as follows from the text of section 3.16.2, which describes the functions of ARN during emergencies, the latter is actively involved in emergency mitigation as a participant (coordinator - practically, manager) along with other institutions. It seems that imposing on the regulatory body the aforementioned functions of emergency mitigation does not fully meet the requirements of the IAEA Standard GS-R-1, prescribing that in case of emergencies the regulatory body shall act as an advisor to the government and competent authorities with respect to the issues of nuclear safety and radiation protection. The active involvement of the regulatory body in performing the nuclear safety and radiation protection tasks along with organizations-performers, which shall bear the main responsibility for these activities, could result in the transfer of such responsibility to the regulatory body, which is unallowable, because that, in turn, causes the factual decrease of safety level. Please provide your comments to the above issue.

The ARN must comply with the National Law No. 24804, regulated by Decree 1390/1998. This is in accordance with the requirements established by the IAEA (Requirement 7, 8, 17, 20 and 21, GS-R Part 1, IAEA), and it should be noted that in Argentina, the competent authority in radiological and nuclear issues is ARN.

Special attention has been paid by the Regulatory Body, starting from the changes established in the mentioned Acts and Ordinances to conduct all the civil organizations and Security Forces in the area included in Nuclear Emergency's Plans. In particular the operational capacity of the Regulatory Body in the management of the implementation of actions to protect the members of the public in the surroundings of the nuclear power plants from the radiological consequences is currently checked by carrying out emergency exercises. The Regulatory Body, in addition to its main role as head at the Emergency Control Centre for off-site consequences, performs the nuclear and radiological assessments, the radiological protection of intervening teams and the environmental monitoring. Representatives of all the intervening organizations (as established in the Emergency Plan) integrate such command and the Regulatory Body coordinates the response teams from civil organizations, (Fire fighter brigades, Civil Defence, etc) security forces (Police, Gendarmerie and Coast Guard) and military institutions (Army, Navy and Air Force). Therefore, this Response System does not affect the safety level.

N° 69

CNS-REF.-ART.: 17

PAGE OF REPORT: 91

CHAPTER OF NAT. REPORT: 3.17.4.2.

In Sect. 3.17.4.2. it is stated that due to the fact that the CAREM-25 prototype will not supply electrical energy to the national grid at its initial stage the implementation of a programme focused on design evaluation and optimization will be facilitated. Are there plans to extend the assessments in case the prototype will be connected to the grid in the future?

The licensing process of the CAREM-25 prototype follows different stages of the project, requiring specific safety analysis to authorize to go from one hold-point to the next one. The project first stage is aimed to assess the safety as well as the neutronic and thermohydraulic behavior of the prototype. This might imply that, for example, a more detailed instrumentation could be preferred even at expenses of the plant availability.

The connection with the grid to be considered in the next stage would imply an update of the safety analysis that would require a new revision by the regulatory authority.

N° 70

CNS-REF.-ART.: 17

PAGE OF REPORT: 101

CHAPTER OF NAT. REPORT: 3.17.5.

See also Annex IX.2.12. Annex IX.2.12. reports an operational event of a dust storm. Has this been incorporated into siting re-evaluations of external hazards?

As was mentioned in the Report, corrective actions have been taken to avoid consequences in plant operation when some systems were not working according to they were designed and to avoid its recurrence. This event was an exceptional occurrence (see IRS N° 8085) that affected service water system and should be studied as a contributor to the initiating event "total loss of low pressure service water". Therefore, it has not been considered to incorporate this event into sitting evaluation as a new external hazard.

N° 71

CNS-REF.-ART.: 17

PAGE OF REPORT: 99

CHAPTER OF NAT. REPORT: 3.17.2.

Argentina presents preliminary versions of a new regulatory standard and of two new Regulatory Guides. Could Argentina present the main changes in the new rules compared with the earliest regulations? Could Argentina specify the references of IAEA's safety documents used as a reference for these new rules?"

There are not main changes in the preliminary version of the new Regulatory Standard AR 10.10.1. "Siting of Class I Nuclear Installations" compared with present regulations of the ARN. As mentioned in paragraph § 3.17.2. the new standard will essentially include the general radiological safety requirements already in force for the site evaluation of a NPP not as a formal standard but as specific regulatory requirements imposed to Licensee. Accordingly, the preliminary versions of the two new regulatory guides related to the new standard AR 10.10.1. include considerations on matters already taken into account by the ARN in the evaluation of sites for nuclear installations in Argentina.

IAEA's safety documents used as a reference for these new ARN's regulatory documents are: "Site Evaluation for Nuclear Installations Safety Requirements" (IAEA Series N° NS-R-3) and its related safety guides.

N° 72

CNS-REF.-ART.: 17

PAGE OF REPORT: 101

CHAPTER OF NAT. REPORT: 3.17.4.2.

The report states that siting studies are under execution"" Could Argentina give details about:
- the method used for this assessment? - the main conclusions of the evaluation?

As it is mentioned in the National Report the site proposed for CAREM-25 is the same as Atucha I and Atucha II NPPs. CNEA is using the NRC RG 1.70 as a guide for the content of the PSAR and ARN is using the NUREG 0800 as a guide for the assessment.

The preliminary main conclusion of the evaluation is that it is a well known site with adequate characteristics to build the CAREM.

Nº 73

CNS-REF.-ART.: 17

PAGE OF REPORT: 103

CHAPTER OF NAT. REPORT: 3.17.6.

The report states that "the country complies with the obligations imposed by the Article 17". Could Argentina give details about the application of the fourth item of Article 17 of the Convention?

Along the time Argentina has signed with its neighbor countries several agreements, protocols of understanding and joint statements related to the pacific use of nuclear energy. These documents include the exchange of information.

As an example it could be mentioned the Protocols for Early Notification and Assistance in cases of Nuclear Accidents and Radiological Emergencies. These protocols establish *"Cooperate, even through the exchange of technical information in order to establish the measures and procedures which contribute to prevent or mitigate the damages coming from a potential nuclear accident or radiological emergency"*.

Nº 74

CNS-REF.-ART.: 17

PAGE OF REPORT: 103

CHAPTER OF NAT. REPORT: 3.17.5.2.

Were the consequences of a dam failure for the cooling water supply of the NPP at the Embalse site considered during the re-evaluation described in Sect. 3.17.5.2. of the report? What were the results of the re-evaluation and the original evaluation regarding this issue?

The lake water intake of the emergency water supply system (EWS) is located at the river-bed which is dammed to form the Embalse artificial lake as was considered in the original plant design. Therefore, in a dam failure case, the feedwater to the EWS will be assured because it will come from the river's tributaries.

Anyway, the results of the seismic re-evaluation carried out in 1986 (considering a horizontal acceleration of 0.26 g and the spectrum used by the NRC) showed that the dam would accompany the ground movements without suffer structure failures and, even if such failures occur, they would take place near the top of the dam assuring the minimum level for EWS pumps operation. This behaviour is expected mainly because of the type of dam. Even when there is still not a specific re-evaluation for the new seismic data, the expert judgements have corroborated the mentioned results.

Nº 75

CNS-REF.-ART.: 17

PAGE OF REPORT: 103

CHAPTER OF NAT. REPORT: 3.17.5.2.

For the Embalse site extensive geophysical assessments have been performed during the re-evaluation described in Sect. 3.17.5.2. Have there been new insights regarding possible seismic sources (faults) in the surroundings of the site?

Field investigations of local faults in the vicinity of the Embalse NPP were undertaken in 2008–2010 as part of the overall seismic re-qualification effort. As a result of this work it has been possible to finalize different credible area source models that better reflect the distribution of earthquakes than previously existing models. The results of the fault study in the field also have made it possible to assign source parameters for the individual faults that reflect opinions among the different experts that took part in the field work.

The field study focused on three geologic structures with the potential to be active faults: the Comechingones fault as exposed about 55 km west of the plant, the Sierra Chica fault passing within a kilometer of the Embalse NPP, and the Elevación Pampeana / Despeñaderos lineament, located about 25 km from the plant site. The main result of the study is that the Sierra Chica fault, the one closest to the site, is either not active ("active in a geological sense"; i.e.: the fault appears to exhibit movement, but so slowly that the probability that it could affect a man-made structure during an

expected service life is low). The observations also allow for the conclusion to be reached that surface rupture beneath the Embalse plant is not an issue, regardless of how close the Sierra Chica fault might be.

The above results are contained in a report recently submitted to NA-SA by the plant designer, AECL, who was responsible for the overall seismic hazard assessment of the plant.

Nº 76

CNS-REF.-ART.: 18

PAGE OF REPORT: 108

CHAPTER OF NAT. REPORT: 3.18.3.3.2.

Referring to a question on the last report, could you comment on the status of the AECL-analysis about the risk assessment of sump clogging and concluded measures?

AECL performed the following activities related to the assessment of the plant for the strainers design:

1-Information Processing:

- 1a- Input data for the development of tasks were collected: lay-out of the reactor building, lay-out plans, specification of the pipes insulation, specification of the paint inside the reactor building, technical specifications of Safety Injection Pumps (SIP), lay-out of suction and discharge pipes in the SIP building.
- 1b- To assess the requirements for filtering (mesh size of the filter, i.e. what size of particles or fibers can afford not to be retained by the filter): characteristics of the elements found along the paths of injection (injection pipe in the primary, valves, orifice plates, geometric characteristics of the fuel, etc.) that could be blocked by particles or fibers downstream of the SIP.
- 1c- Accumulation of water in the sump, water temperature and containment pressure during postulated accidents for the design (available level for the strainer).

2-Debris Bypass and Thin Bed Head Loss Testing: in order to define the mesh size:

Considering that from point 1.b it was required a special mesh size (unusually small), some alternatives were analyzed and it was decided to use another fine mesh of normal size. An experiment is carried out to optimize the mesh design (which is required to minimize the hydraulic resistance).

3- Coating System – AECL personnel site visit – coating qualification:

As an overall conclusion it could be said that painted surfaces were found in good condition excepting the sump. As a consequence, new procedures and materials were used to re-paint the sump. This task is already finished.

4-Debris Assessment:

- 4a-Identification of critical break size and locations;
- 4b-Determination of debris volume and types;
- 4c- Identification of debris flow path and postulated transport debris;
- 4d- Calculation of debris volume retained on the strainer;
- 4e- Calculation of the net positive suction head available for the ECC pump and;
- 4f- Chemical effect assessment.

5-Finally, the conceptual design was defined and a walk-down was performed in order to evaluate possible problems for the strainers erection.

6- Technical Specifications:

The design was finished. All the characteristics were defined. A report was delivered to NA-SA.

7-An Hydraulic Analysis was performed.

8- Stress and Seismic Analysis was also performed.

9- Reduced-scale test plan:

These tests were performed last December.

The strainers are being constructed in Canada and should be delivered in next July.

Nº 77

CNS-REF.-ART.: 18

PAGE OF REPORT: 104 / 108

CHAPTER OF NAT. REPORT: 3.18.2.

See also page 106 Article 18(ii) requires that: “The technologies incorporated in the design and construction of a nuclear installation are proven by experience or qualified by testing or

analysis". Bearing in mind that CNA II is of hybrid Konvoi and PHWR technology, to what extent has it been treated as a prototype, specifically in the safety analysis and also in special testing of features in the commissioning programme?

CNA II has CNA I as predecessor in PHWR technology. The 35 years of CNA I operating experience was taken into account in CNA II design. Additionally CNA II design was improved with the use of Konvoi technology. Therefore, CNA II is an improved version of CNA I and is not a hybrid.

The Final Safety Analysis Report in the Accident Analysis Chapter, 12 specific postulated initiating events (PIE) are included. Also additional specific PIEs for the PSA were required.

The commissioning programme is under review. Special testing to verify the neutronic and thermohydraulic design as well as the system's I&C is one of the important issues of the commissioning programme.

Nº 78

CNS-REF.-ART.: 18

PAGE OF REPORT: 114

CHAPTER OF NAT. REPORT: 3.18.3.

The report provides a table with the progress of the main construction items (civil, piping, mechanical equipment, HVAC, electrical equipment and I&C). Can you provide an update of the status of the construction of these items and an update of the schedule for the start of operation of CNA II?

The construction will be finished during September 2011 and the Commissioning of the plant will be started. The start of the operation could be defined during the commissioning.

Nº 79

CNS-REF.-ART.: 18

PAGE OF REPORT: 117

CHAPTER OF NAT. REPORT: 3.18.3.5.

The report states that different technological alternatives are under study for the construction of the fourth nuclear power plant in Argentina. Can you provide additional information regarding what technologies are being considered?

It is being considered the enriched uranium-light water technology.

Nº 80

CNS-REF.-ART.: 19

PAGE OF REPORT: 12

CHAPTER OF NAT. REPORT: 3.19.6.

The report states that "the operator is performing its technical and safety assessment". Could Argentina specify if the operator's technical and safety assessment is assessed by the Regulator?"

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 ARN requirements or as a demand of the Licensee itself. ARN controls the safety level by means of audits and assessments of the studies carried out by the licensee or verifying the results by performing its own analysis.

Safety assessment and verification constitute the basis and technical support of the regulatory control.

Nº 81

CNS-REF.-ART.: 19

PAGE OF REPORT: 122

CHAPTER OF NAT. REPORT: 3.19.7.

The report states that "Regulatory Standard AR 3.9.2. sets the basic criteria concerning definitions, event communication modes to the ARN, and event analysis." Could Argentina present a short list of the criteria used for event reporting to the IRS database?

Is necessary to clarify that Regulatory Standard AR 3.9.2. related to Communication of Relevant Events in NPPs, refers to the communication from the Licensee to ARN, in case of a relevant event occurrence and is not related to ARN event communication.

Criteria used for event reporting to the IRS database are based in IRS manual. As examples:

- Importance of the event to be share as operating experience
- Recurrence of the event or misuse of international operating experience feedback
- Events related with design changes or component aging

Nº 82

CNS-REF.-ART.: 19

PAGE OF REPORT: 121 / 122

CHAPTER OF NAT. REPORT: 3.19.5.

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. There is no evidence that there are special procedures such as symptom-based emergency operating procedures (SB EOPs). Is it true that all emergency operating procedures are only event-based? Are there any plans to harmonize EOPs with severe accident management guidelines (SAMGs) in the future?

The EOPs are based on events but they are symptom oriented. This means that they are event ordered orienting the operator through the corresponding symptoms that have to find in order to confirm the accidental scenery diagnosis. Severe accident management guidelines are being developed for Atucha I and Embalse in the corresponding SAMP framework. More details are given in section 3.14.3.1.2., page 73 of the Argentinean national report.

Nº 83

CNS-REF.-ART.: 19

PAGE OF REPORT: 122

CHAPTER OF NAT. REPORT: 3.19.8.

Procedures in accidental conditions are applied by different operation shifts during theoretical exercises as well as during simulator practices. Does it mean that only the main control room shift personnel are trained for accidental conditions and only exercises on the simulator are used for this purpose?

The operating personnel are annually trained in a full scope simulator considering different postulated initial scenarios in order to apply the EOPs. Also receive this training the personnel involve in the EOPs preparation that belongs to the plants Nuclear Safety Department.

On the other hand, for each plant, the emergency plan foresees an annual emergency exercise used as a tool to train the plant personnel to response in different accidental scenarios.

Nº 84

CNS-REF.-ART.: 19

PAGE OF REPORT: 119

CHAPTER OF NAT. REPORT: 3.19.2.3.

See also 3.18.3.4. CAREM Reactor Prototype Page 114 and 3.19.2.3. CNA II- Initial Authorization to operate Page 119 The report refers to the impending commissioning of CNA II. It also refers to a prototype NPP, CAREM. Given that there has been no commissioning of NPP for many years in Argentina, some more detail on arrangements for commissioning would be useful. For example, what arrangements are in place in the licensee to commission the plants? What organizational structure has been or is to be enacted? What regulatory activities are planned to oversee commissioning and what permissioning activities are underway or planned by ARN?

a) Arrangements and organizational structure of the Licensee:

In order to complete the Atucha II NPP (CNA II) construction a Design Authority is clearly stated though NA-SA resolution 72/09. It determines responsibilities into three levels of the organization: NA-SA (the owner and licensee), the Atucha I-II plant manager (Primary Responsible), NA-SA UG CNA II (sector of NA-SA responsible for finalizing the basic and detailed engineering, and the CNA II construction). The methodology for the control of design changes has three levels, a first level that performs the basic and detailed engineering and two review levels: The Review Committee for Relevant Design Changes acts in the UG CNA II and an independent Technical Review Committee acts afterwards in NA-SA level (licensee).

The Licensee constituted a “responsible organization” for conducting the commissioning process and an “Ad Hoc” Committee responsible for the verification and acceptance of the results from each one of the different commissioning stages and the authorization of the next one.

Arrangements that the licensee is performing related to CNA II commissioning follow the recommendations of IAEA Nuclear Engineering Series N° NG-T-2.2 and, among them it is possible to mention:

- The Licensee defined key positions of commissioning of Atucha II that are required to be staffed by experienced persons from Atucha I.
- Specific training materials are used including a “full scope” simulator.
- On the job training during commissioning for experienced and new staff
- Participation of the experienced persons in the review of the commissioning programme and procedures.
- The staffing plan for the new NPP project anticipates recruiting additional staff for Atucha I.

NA-SA has proposed a new organization for Atucha site after Atucha II commissioning. The commissioning organizational structure is in line with the organization mentioned before. In some activities as maintenance the same personnel will be used for both CNA I and CNA II, others activities as operation will have different shift operators.

All the mentioned activities had the agreement of ARN.

b) Permissioning activities underway or planned by ARN:

Currently the plant activities are covered by a construction license, which shall include the preliminary tests. Once fulfilled satisfactorily these preliminary tests, ARN would issue a commissioning license in case the Licensee completes adequately the required mandatory documentation. This License authorizes the fuel and moderator load, first criticality and commissioning from low power to full power. The commissioning programme is under review. Special testing to verify the neutronic and thermohydraulic design as well as I&C is one of the important issues of the commissioning programme.

ARN will supervise and control all the tasks related with the preliminary tests (cold and hot tests) as well as nuclear commissioning activities.

Regulatory activities related with the mentioned tasks will be plan and implemented by an “ad-hoc” organization which is being defined. Besides, it is foreseen that reputable TSOs advise ARN during CNA II commissioning.

In relation with the preliminary tests, ARN will focus the regulatory effort on a set of systems which have been selected according to their safety relevance.

ARN will control and assess the activities related with all the nuclear commissioning steps from first criticality to full power operation through inspections, safety assessments and audits.

N° 85

CNS-REF.-ART.: 19

PAGE OF REPORT: 120

CHAPTER OF NAT. REPORT: 3.19.3.2.

It is understood that the operational limits and conditions for NPPs in Argentina are contained in the Policies and Principles Manuals. With regard to this manual, could you provide following information:

- i) Is there any practice to revise the manual based on operating experience and new findings?**
- ii) What is the role of ARN in the process of revision and whether any prior approval is required for effecting changes?**
- iii) What is the experience of NPPs in meeting the requirements of the Manual? If there are any instances of non-conformance, how are these dealt with?**

i) As it is established in the corresponding Operating License, the plants Policies and Principles Manuals (P&P) have to be periodically updated. Actually operating experience and new findings from other sources like PSA updates lead to a P&P review.

Owing to P&P is part of the mandatory documentation any change in it has to be approved in several Licensee internal instances.

- Plant Engineering Sector proposes the modifications and prepares a report that includes a safety assessment;
- Headquarters engineering group produces a review (considering internal or external technical advice);

- The output from the two previous steps is discussed within the Internal Plant Review Committee which acts as the plant manager advisor.
- The plant manager sends the set of documents to the Technical Review Committee which advises NA-SA general manager. Finally, when it is approved it is sent to the Regulatory Body and;
- Once the Regulatory Body authorizes the modification the P&P new version is approved.

ii) According to what is established in the plants Mandatory Documentation, it is required that all modification proposed for this documentation which has a significant influence on the plant safety must be evaluated by the Technical Review Committee and authorized by the Regulatory Body prior to be implemented. Therefore, is a condition to authorize a P&P change, that the Regulatory Body performs an independent assessment considering its safety relevance; it means that it is assessed and analyzed all the documentation related with the proposed change.

iii) As a normal and routinely practice the P&P requirements are met. However, in exceptional cases of a P&P non conformance, a safety evaluation could be performed by the plant technical support group which results are reviewed by the Internal Plant Review Committee that advise the plant manager in the decision making process. This situation must be immediately informed to the Regulatory Body. Some past examples refers to AOTs or time between tests.

As were mentioned in the Argentinean National Report, any violation to what is established in the Mandatory Documentation makes the Licensee liable to be sanctioned according to the in force Sanction Regime.

Nº 86

CNS-REF.-ART.: 19

PAGE OF REPORT: 120

CHAPTER OF NAT. REPORT: 3.19.3.1.

Is there any program for surveillance of seismic category-I structures?

Emergency Water Supply (EWS), Emergency Power Supply (EPS) and the seismically qualified portion of ECCS apart from active equipment routine tests are included in a surveillance program as described below:

- EWS: pipes thickness measurements by ultrasonic test, visual inspection of valves and components, RX for welded joints and analysis of the diesel engine fuel;
- EPS: visual inspections of components, fuel tank ultrasonic inspection, analysis of the diesel engine fuel;
- ECCS: visual inspections of piping, tanks, valves and supports, complete piping ultrasonic inspection, RX for welded joints, magnetic particles inspection and dye penetrate testing of valves stud, visual inspection and ultrasonic inspection of heat exchanger plates and routine chemical analysis of system water.

Nº 87

CNS-REF.-ART.: 19

PAGE OF REPORT: 124 / 125

CHAPTER OF NAT. REPORT: 3.19.8.2. / ANNEX IX

In the German NPP Krümmel due to a fire of a transformer fire gases could enter the control room. Connected to this event several measures were performed in German power plants. Was this event part of the Argentinean OEF process? Were measures taken to prevent such an event?

This event has been considered to be included in the external operating experience for application in Embalse and Atucha I and II. At present it is still under analysis. However, a preliminary revision showed the existence of relevant plant layout and design differences among Krummel and the Argentinean plants.

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 7 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 station contains a reactor of the pressure vessel type, and is fuelled with natural uranium as well as slightly enriched uranium; it is heavy water moderated and cooled (being of the PHWR type); it is periodically refuelled on power.

Besides, the plant also has the big systems which characterize the classic electric power generating plants, steam turbine and electric generator, as well as components, equipment and sub-systems required for the functioning of the big systems located at its "nuclear" and "conventional" sections.

Figure 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 station 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 station service transformer.
- (13) Generator transformer.
- (14) Off-site system transformer.
- (15) Secondary Heat Sink.

In the CNA I, a clear physical separation exists between the nuclear and conventional sections of the plant. The reactor building, the reactor auxiliary building and the fuel storage building constitute the "controlled area" in which all systems assigned to the nuclear section are installed. In this way the potential radioactivity is limited to defined regions. There is only one controlled access to the "controlled area".

The reactor building is linked with the reactor auxiliary building through a personnel airlock and this auxiliary building is, in turn, linked to the fuel storage building through an underground tunnel. The rest of the buildings are located in the "conventional" section of the nuclear power plant.

The reactor building ([Figure 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 station 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 emergency diesel generator and the high-voltage station service transformer.

The main steam lines coming from the reactor building enter the turbine building along the shortest route leading to the area of the high pressure casing of the turbine, where the main steam flows through the steam strainers into the high pressure turbine. Vertical moisture separators are installed on both sides of the high-pressure casing.

The turbine operating floor is clear of off pumps and piping so that floor space is available to set down components removed when the turbine generator unit is opened up 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 smoke 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.

III.1.3. CNA I Main Systems

In what follows the main safety and process systems that are part of the station, are briefly summarised.

III.1.3.1. Reactor

The reactor (*Figure III.1-5*) is of the pressure vessel type, natural and slightly enriched uranium fuelled and heavy water moderated and cooled. The bulk thermal power is 1179 MWt.

The reactor core is approximately cylindrical in shape and consists of 252 enriched uranium fuel assemblies located in the same number of coolant channels. The fuel assemblies are bundles of 36 closely packed fuel rods which are arranged in 4 concentric rings having 1, 6, 12 and 17 fuel rods each, plus an additional structural rod located in the external ring. Each fuel rod consists of a stack of uranium dioxide pellets enclosed by a thin walled zircaloy 4 canning tube, which is both gas and pressure tight. Each fuel assembly, together with the filler body and the closure plug, forms the fuel bundle column. The coolant channels are arranged vertically in a trigonal lattice within the moderator tank. Each coolant channel contains one fuel bundle column.

The coolant channels are surrounded by the moderator, which is enclosed in the moderator tank. For reactivity reasons, the moderator is maintained at a lower temperature than the reactor coolant. This is accomplished by the moderator system, which extracts the moderator from the core, cools it down in the moderator coolers, and feeds it back into the core.

The heat removed from the moderator is used for pre-heating the feed-water. This is one of the reasons for the high net efficiency of the power plant.

The reactor coolant system and the moderator system are connected by the pressure equalisation openings of the moderator tank closure head. Therefore, the pressure differences in the core are comparatively small, which results in thin walls for the reactor pressure vessel internals. This allows a very high burn-up to be attained. Furthermore, the connection between the reactor coolant system and the moderator system permits the use of common auxiliary systems to maintain the necessary water quality. The number of auxiliary systems can therefore be reduced to a minimum.

In order to control reactivity, and thus the power output of the reactor, various methods are applied. The reactor contains 24 "black" (absorbers made of hafnium) and 5 "grey" (steel) control rods. The control rods are used to control reactivity and power distribution, to compensate the build-up of xenon poisoning after a reactor power reduction, to provide damping of azimuthal xenon oscillations, and to shut down the reactor.

In addition to the control rods, reactivity control is provided by the boric acid dosing system. The injection or extraction of boric acid serves to compensate slow reactivity changes due to burn-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 fixed 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 then to the service cooling water system.

During emergency core cooling the moderator serves as a high-pressure core re-flooding and cooling system. The emergency core cooling position is similar to that of the residual heat removal, but additionally, water is injected into the hot legs of the reactor coolant loops and into the upper plenum of the reactor pressure vessel. The residual heat removal chain connected to the moderator coolers during emergency core cooling is the same as during residual heat removal.

An essential feature of the moderator system together with the residual heat removal system is the allowing of the hot shutdown condition of the reactor to be maintained for as long as required, or the cool-down at a pre-set gradient, as well as emergency core cooling without main steam blow-off and thus without an additional heat sink.

All systems of the residual heat removal chain are of a consistent two-loop design. The residual heat removal system acts as a barrier between the active moderator and the service cooling water and prevents the escape of radioactivity into the service cooling water in the event of leakages in the moderator coolers.

III.1.3.3. Refuelling system

The slightly enriched uranium reactor makes it possible and desirable, with a view to obtaining a high burn-up, to shuffle and replace the fuel assemblies during power plant operation. The refuelling procedure is carried out by a single refuelling machine. The fuel assembly transport system is located in the reactor building and in the fuel pool building. The main items of the fuel transport system are: refuelling machine, tilter with supporting structure, fuel transfer tube, fuel pools, and the corresponding auxiliary systems and maintenance installations.

The refuelling procedure is fully automated and monitored from the control room.

The refuelling machine is moved from a maintenance position in the refuelling machine maintenance room, by remote control, to a previously selected coolant channel position in the reactor well in which

the machine is centered. The seat-on seal is pressed on to the coolant channel closure head by the dead weight of the refuelling machine to form a watertight seal between the machine and the coolant channel. Pressure equalisation takes place between the refuelling machine and the reactor before opening the isolation valve of the refuelling machine and opening the coolant channel closure. Following this, the fuel bundle column is withdrawn into an empty position in the refuelling machine magazine. The magazine is then rotated in such a way that a fuel bundle column with a partially burnt-up fuel assembly or with a new fuel assembly is positioned above the open coolant channel. This fuel bundle column is lowered into the coolant channel position and the coolant channel closure is locked again. After closing the isolation valve of the refuelling machine a check for leak-tight closure is performed. Then the refuelling machine is removed from the reactor pressure vessel and positioned above the vertically arranged tilter. The tilter has the following functions in the indicated sequence:

- Take-over of the fuel bundle column with the spent fuel assembly.
- Removal of the decay heat by cooling with heavy water.
- Drying and cooling the spent fuel assembly with gas.
- Flooding and cooling of the tilter with heavy water.
- Tilting to the horizontal position and connecting with the fuel transfer tube.
- Transfer of the fuel assembly into the fuel transfer tube.

When a new fuel bundle column is transported from the fuel pool building into the tilter via the transfer tube, and later from there into the refuelling machine, the process of cooling and change of cooling medium takes place in the reverse order.

The task of the fuel transfer tube is to establish a connection between the tilter within the reactor building and the tilting device in the fuel pool building, while both components are in the horizontal position.

The tilting device takes the fuel bundle column from the fuel transfer tube and swivels it from the horizontal into the vertical position.

Besides the main components mentioned above, important auxiliary systems, tools and maintenance and service equipment necessary inside the fuel pool building are provided.

The great advantage of the refuelling system is that it handles the fuel assemblies with only one refuelling machine and that it has one seal ring for each coolant channel. A silver-clad seal ring with good material flow properties is used as sealing material. In this established design, the coolant channel seals are almost perfectly tight.

Using the gas drying and cooling circuit in the tilter, the heavy water humidity is completely removed from the tilter and then recovered by cooling and condensing the extracted gas stream. This is one of the reasons for the very low heavy water -in Atucha I Nuclear Power Plant. With this fuel transport system, it is also possible to transport semi-burnt fuel assemblies from the fuel pool building to the reactor.

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 station auxiliary service requirements by means of one high-voltage station service transformer.

The high-voltage station service transformer or the off-site system transformer feed into two separate medium (each 6.6 kV) high voltage bus sections, to which the large auxiliary loads and the transformers for the low voltage switchgears are connected.

If the station service power system fails, certain equipment (pumps, etc.) are needed to remove residual heat and to run the plant into safe conditions. This equipment must remain in operation or has to be put into operation and must therefore be supplied with emergency power.

The emergency power system is like the other safety equipment divided into redundant separate trains. Under normal operating conditions, the auxiliary switchgears of the auxiliary power system feed the emergency power system. To avoid loss of power in case the auxiliary power system fail, each of the redundant trains in the emergency power system is equipped with a quick-starting diesel set.

CNA I 's electric system may be divided into two main subsystems: the offsite power system and the onsite power system.

The offsite power system is constituted by the 220 kV and 132 kV transmission lines connecting CNA I with two sub-stations belonging to the national electric grid.

With the generator load-breaker in the "off" position the station service power for "start up" and "shut down" of the CNA I can be drawn from both 220 kV grid and 132 kV grid. Upon simultaneous failure of

the main grid and the turbine generator set the 132 kV grid provides power for shut down operation of the plant down to the "hot-subcritical" condition.

The onsite power system consists, in turn, of two subsystems: the auxiliary power system ("normal system") and the emergency power system.

The auxiliary power system provides power for the loads of the nuclear power plant, which are necessary during normal operation, start-up and shutdown operation. It is subdivided into two trains {6.6 kV buses BA and BB} which are supplied normally by the high-voltage station service transformer. This transformer is fed either from the generator or from the 220 kV grid via the generator transformer. For shutdown operation or after loss of the normal power supply grid and generator, it may be fed by the off-site power supply via the off-site system transformer. The offsite power supply system is available via automatic changeover.

The emergency power system provides the power required for safe shut-down of the reactor to maintain it in the shut-down condition, for removal of residual heat and to prevent release of radioactivity during normal operation and accident conditions, and for some loads important for plant availability. It is subdivided into two trains - 6.6 kV uninterruptable BU and BV buses-, which are usually supplied by 6.6 kV buses BA and BB as well as the water turbine driven generator (located in the water turbine building).

During emergency situations only safety related loads are fed. For those situations in which one or more out of the three power supply possibilities before mentioned are available, buses BU and BV continue to be fed by buses BA and BB. In this last case, the condition of uninterruptible voltage in BA and BB is initially achieved by the emergency diesel generators supply.

As the water turbine-driven generator was connected to bus BV before the emergency situation, this turbo-generator continues supplying energy after the emergency signal, during a time period of 40 seconds; during such period, the emergency diesel generators start and they are connected to buses BU and BV.

The emergency diesel generators are three redundant units and it is enough that any two of them are functioning, for the supply of all the loads fed from buses BU and BV. Each emergency diesel generator is, in turn, constituted by different main and auxiliary subsystems, such as the starting subsystem, the lubrication subsystem, etc.

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 safety 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 coolant channels or fuel assemblies	252
Cladding material	Zircaloy 4
Fuel assemblies length	6180 mm
Refuelling	On load
Coolant and moderator	Heavy water

Thermal and Hydraulic Data	
Pressure at reactor vessel inlet	12.2 MPa
Pressure at reactor vessel outlet	11.6 MPa
Coolant channel inlet temperature	264 °C
Coolant channel outlet temperature	303.3 °C
Maximum temperature on the fuel assembly cladding surface	325 °C
Coolant flow in coolant channels	20210 t/h
Average coolant speed in central channel	9 m/s
Mean heat-flux density	67.7 W/cm ²
Average specific thermal power of fuel rods	232 W/cm
Heavy water concentration	99.75 (weight)

Steam and Power Conversion System Data	
Live steam pressure at steam generator outlet	4.46 MPa
Live steam temperature at steam generator outlet	254.9 °C
Live steam flow	1856 t/h
Live steam moisture	0.3%
Turbine rated speed	3000 rpm
Condenser pressure	4.56 kPa
Cooling water inlet temperature of condenser	22 °C
Cooling water flow of condenser	62500 m ³ /h
Generator apparent power	425 MVA
Generator power factor	0.8
Generator voltage	21 kV
Generator transformer rated power	400 MVA
Generator transformer transformation ratio	21 kV / 245 kV
High-voltage station service transformer rated power	35 / 20 / 20 MVA

High-voltage station service transformer transformation ratio	21 kV / 6.95 kV
Generator off-site system transformer rated power	35 / 20 / 20 MVA
Off-site system transformer transformation ratio	132 kV / 6.95 kV

SITE LOCATION

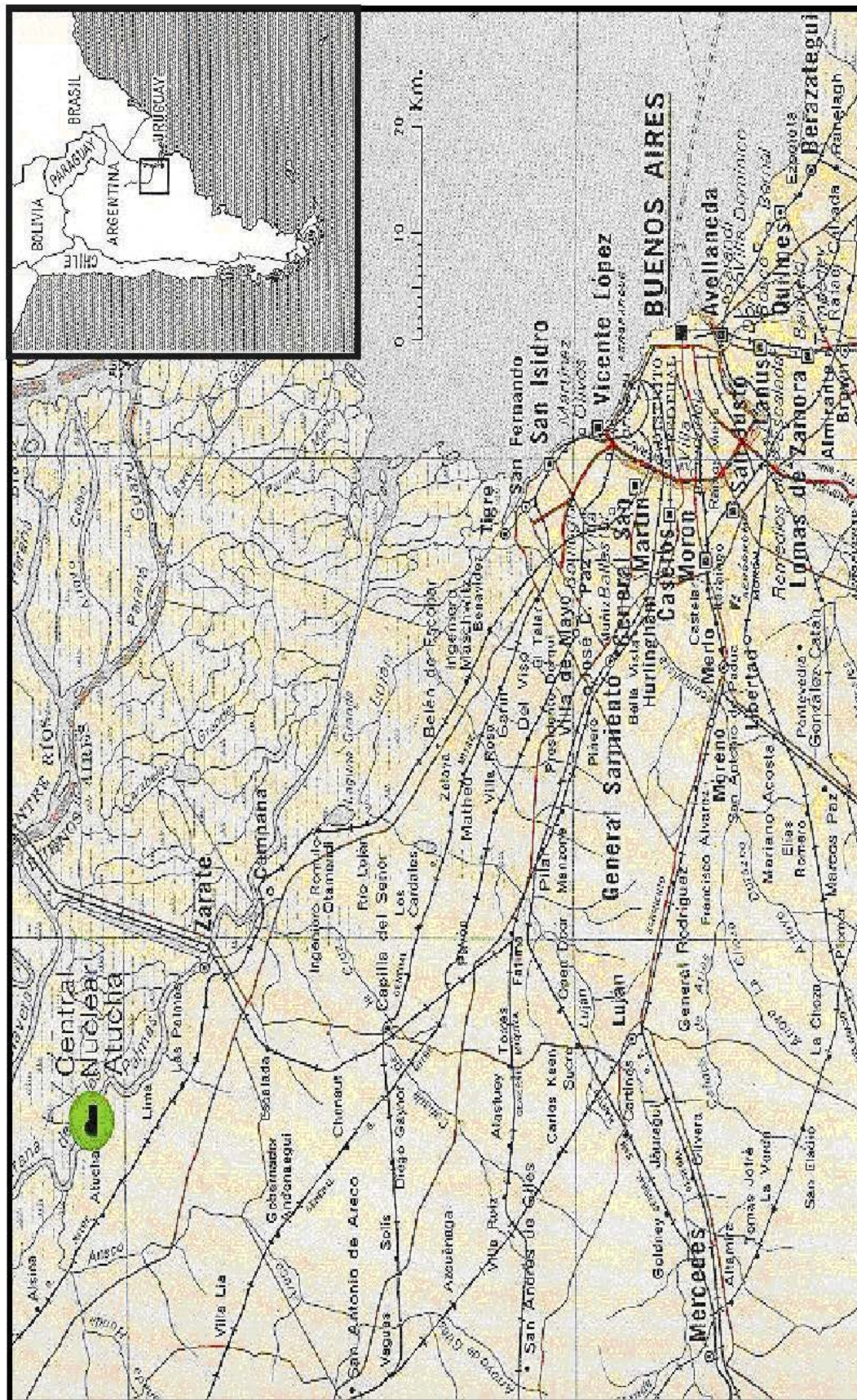


Figure III.1-1 - Atucha I Nuclear Power Plant - Geographic Location

CNA I SIMPLIFIED FLOW DIAGRAM

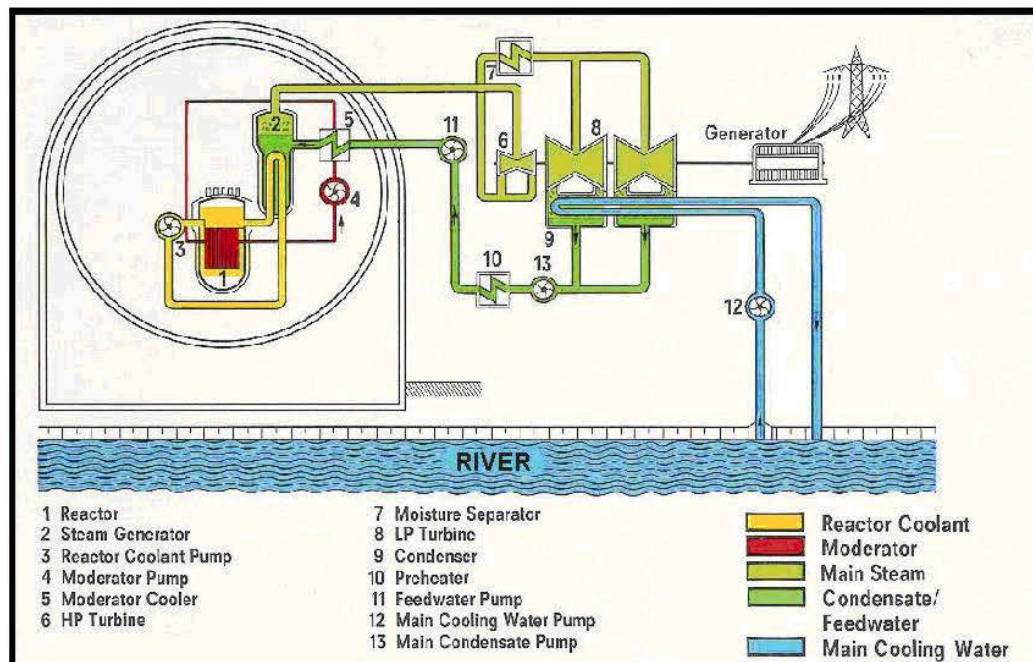
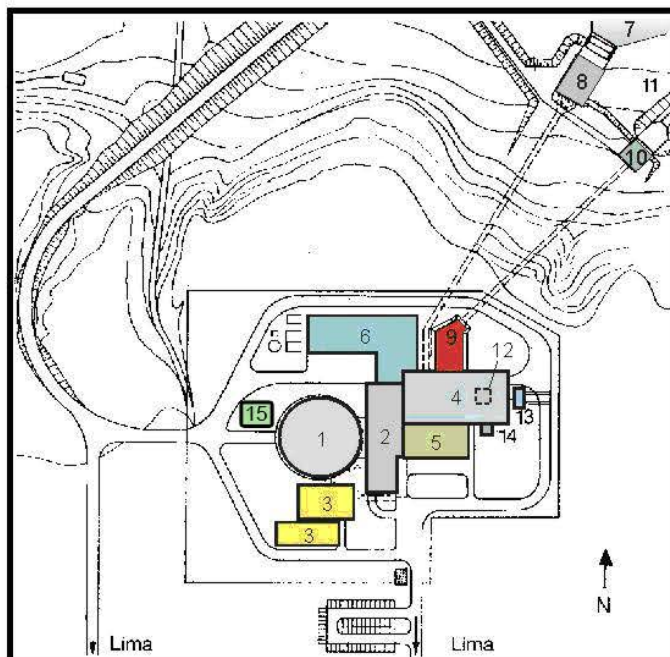


Figure III.1-2 - Atucha I Nuclear Power Plant - Main Systems

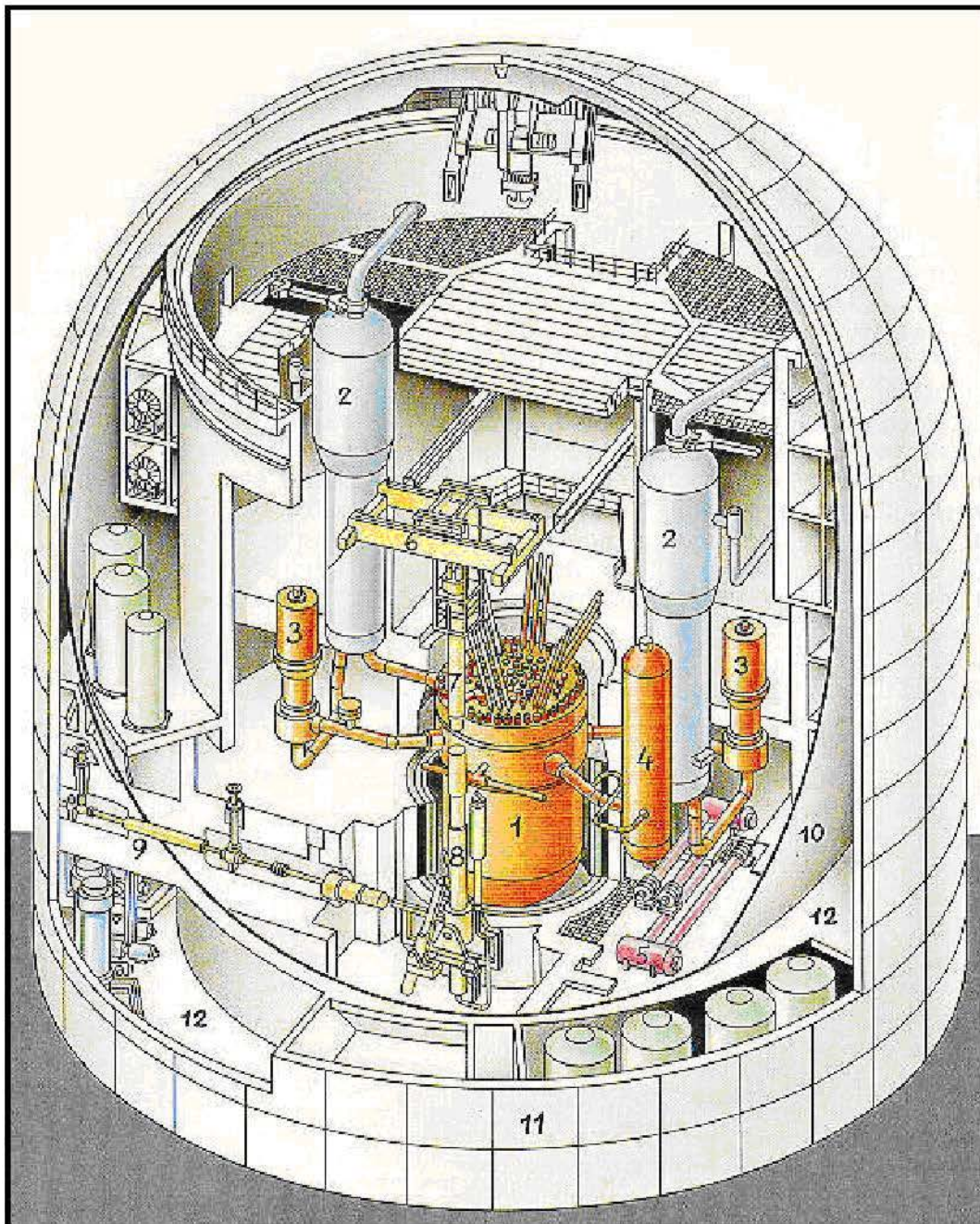


SITE PLAN

- (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 Coding Water Pump House.
- (9) Cooling Water Seal Pit.
- (10) Water Turbine Building.
- (11) Cooling Water Outfall Channel.
- (12) High-Voltage Station Service Transformer.
- (13) Generator Transformer.
- (14) Off-Site System Transformer.
- (15) Secondary Heat Sink.

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

REACTOR PRESSURE VESSEL - INTERNALS

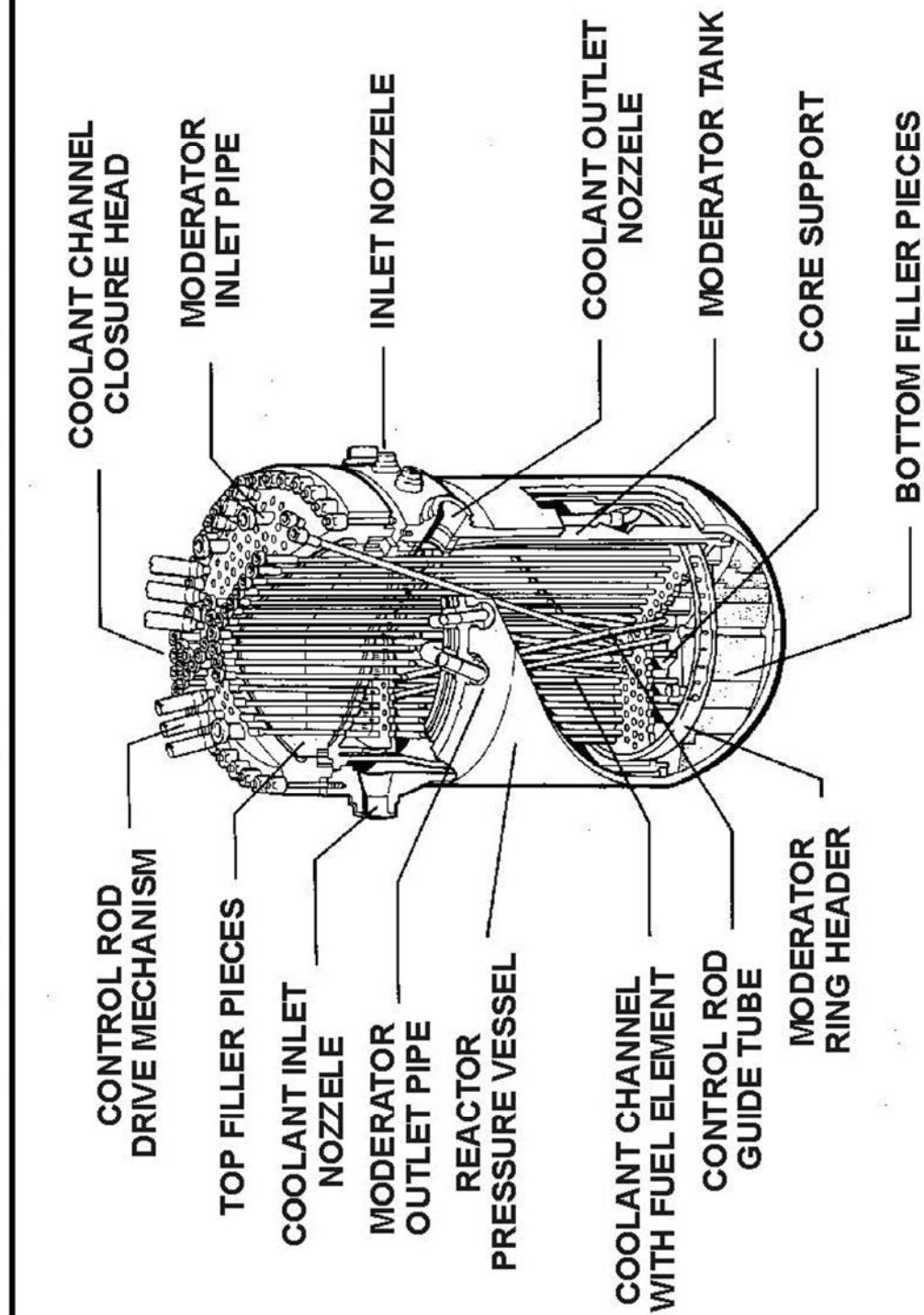
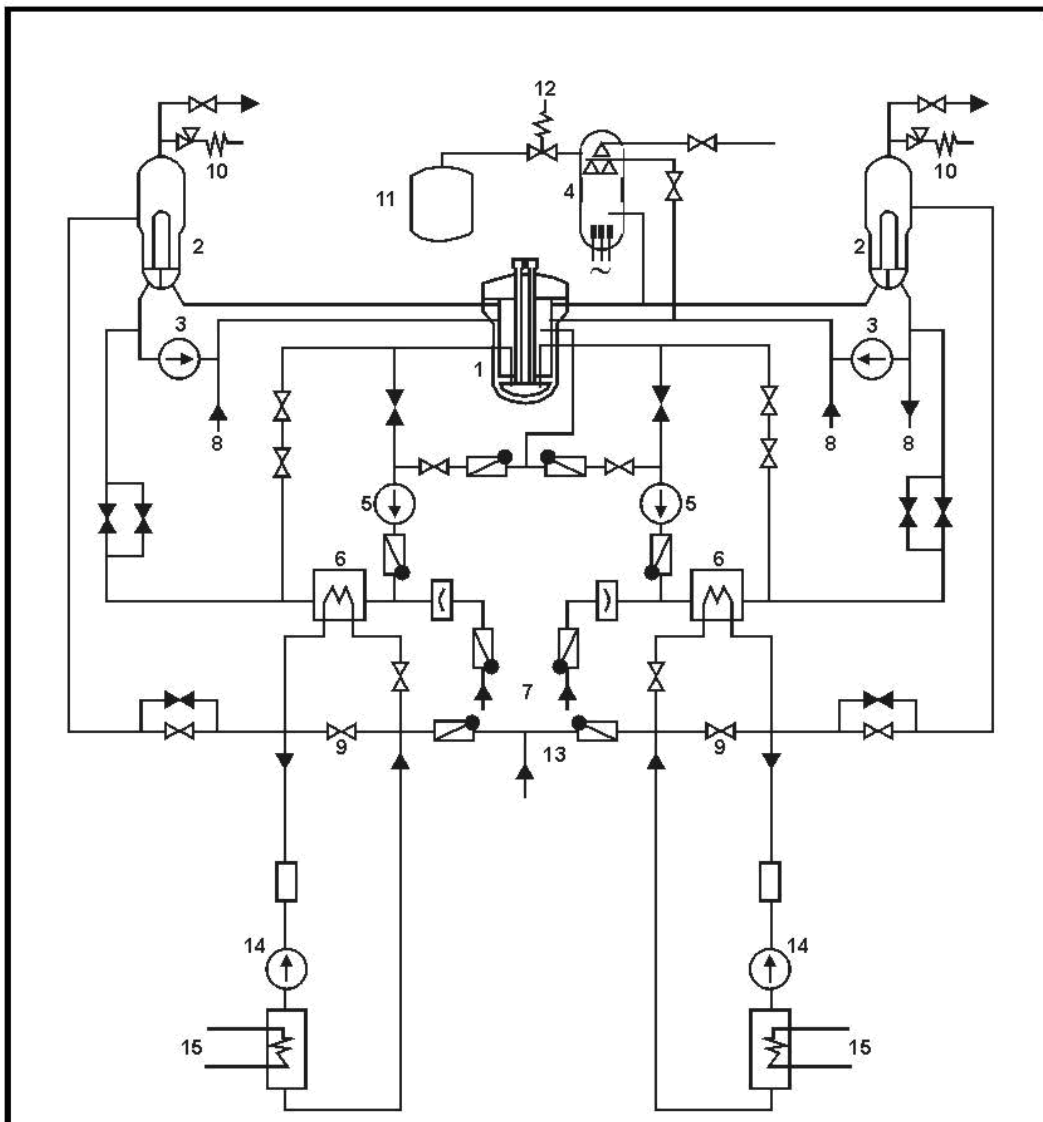


Figure III.1-5 - Atucha I Nuclear Power Plant - Reactor Pressure Vessel and Internals

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*

[illegible]

BS - Off-site System Transformer

BY - Diesel Generators

UB - Water Turbine - Driven Generator

Main Technical Features of the Argentine Nuclear Power Plants in Operation **ANNEX III** 17

ANNEX IV

PRINCIPAL TECHNICAL CHARACTERISTICS OF ATUCHA II NUCLEAR POWER PLANT

IV.1. INTRODUCTION

The CNA II –a PHWR type of 700 MWe– was designed by the Siemens, with the participation of ENACE as architect engineer at the time the project began. CNA II construction license was issued in July 14, 1981. The construction and commissioning process is, since 1994, under the direct responsibility of the State enterprise NA-SA, as the Licensee.

The 700 MWe PHWR makes full use of KWU's experience in the light and heavy water reactor fields and the operating experience of CNA I, a station that has shown excellent operating performance with high rates of availability and thereby furnished proof of its full operating reliability.

The pressure vessel type PHWR implemented in the 700 MWe PHWR is derived from CNA I and the 1300 MWe KWU standard PWR. Thus, the heavy water specific components such as moderator pumps, moderator coolers, moderator valves, coolant channel closures, refueling system, heavy water upgrading columns, etc. used are of almost identical design with CNA I; all other components of the nuclear and conventional part of the plant, almost all main and auxiliary systems and the layout of the buildings are derived from the KWU standard PWR design.

CNA II plant is located 110 km north-west of Buenos Aires city, on the southern bank of the Río Paraná de las Palmas, in 9 km distance of the Community of Lima, Zárate County, Province of Buenos Aires. The plant is located adjacent to the east side of CNA I (*Figure IV-1*).

The station contains a reactor of the pressure vessel type, fuelled with natural uranium (like CNA I, it could also be fuelled with slightly enriched uranium); it is heavy water moderated and cooled. The moderator heat is used for preheating of steam generator feedwater and its temperature for reactivity control of the reactor. The reactor is on-load refueling with a single refueling machine arranged on top of the reactor pressure vessel.

Four redundant trains are installed for every safety system, thus enabling repair work on one train during plant operation. In every conceivable accident condition, the reactor plant can be kept "hot subcritical" or cooled down with the help of the high pressure residual heat removal system or with the emergency feeding of water on the secondary side of the steam generators as in PWR technology.

Figure IV-2 shows schematically the main systems of CNA II.

IV.2. OVERALL PLANT LAYOUT

The overall layout (*Figure IV-3*) and arrangement of the CNA II on the site is governed by the following basic considerations:

- Clear separation of the nuclear and conventional systems.
- Clear energy flow paths.
- Short piping and cable runs.
- Good transport conditions and access for construction, installation and operation.

Buildings and structure arrangement of the CNA II are shown in *Figure IV-3*. As it can be seen from the site plan, the main buildings and structures of the plant are:

- Reactor building, incorporating containment structure and annulus.
- Reactor auxiliary building with heavy water enrichment tower and vent stack.
- Fuel store building, with storage areas for new and spent fuel assemblies.
- Switchgear building, including the plant control room.
- Turbine building.
- Switchgear building.
- Fuel storage building.

- Emergency power and chilled water supply building.
- Main steam and feedwater valve compartment.

CNA II has a clear physical separation between the nuclear and conventional sections of the plant. The reactor building, along with the annulus, reactor auxiliary building and the fuel storage building, constitutes the "controlled area" in which all systems assigned to the nuclear section are installed. In this way the radioactivity which arises is limited to defined regions. There is only one controlled access to the "controlled area".

All pressure retaining components of the nuclear steam supply system such as the reactor, the reactor coolant system, the moderator system and associated equipment are arranged inside the reactor building, which is enclosed by the inner spherical steel containment and the outer concrete shield. The containment structure is designed for the maximum pressure associated with the worst event which has to be taken into account.

A special ventilation system for the annulus ensures that even under accident conditions small radioactive leakages from the containment are retained try charcoal filters, thus preventing any radiation hazards to the environment. The systems necessary for on-load refueling are also housed in the containment structure.

In the lower part of the annulus between the containment sphere and the concrete shield various auxiliary and ancillary systems are accommodated, such as: residual heat removal system, safety injection system, heavy water storage system and components of the reactor cooling system.

The reactor auxiliary building adjoins the reactor building, and surrounds a part of it, thus allowing short connections to the equipment located in the reactor building annulus.

On the upper floors of the building there are active and inactive sanitary rooms, the laundry with ancillary rooms, the controlled access area ventilation system, the radiochemistry laboratory, the areas for radiation protection and the respiration apparatus room. From this area of the reactor auxiliary building there is an access to the reactor sphere via a personnel airlock.

The lower floors accommodate different auxiliary systems, such as: volume control system, heavy water purification and degassing system, heavy water treatment and enrichment system, boric acid and chemical control system and the gaseous, liquid and solid waste processing systems.

The fuel storage building is linked with the reactor building by the fuel transfer system. Personnel access is possible from the reactor auxiliary building.

Inside the building, there are four fuel storage pools, a manipulating pool, a small pool for the spent fuel shipping cask, a new fuel store and the necessary auxiliary equipment. The spent fuel assemblies are transferred from the reactor to the fuel storage pool with the aid of the fuel transport system, consisting of: refuelling machine, tilter, transfer tube, tilting device and manipulating bridge. The fresh fuel assemblies are supplied to the reactor in the reverse way.

The switchgear building has nine floors. They are used as follows:

- Cable ducts.
- Cable basement.
- High voltage switchgear.
- Cable race below D.C. systems.
- Battery, rectifiers, 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.

Access to the switchgear building is from the staff facilities and office building via a personnel passageway. Access to the reactor auxiliary building is at the same level. The personnel passageway between staff facilities and office building/ turbine building allows passage between switchgear building and turbine building.

The off-site power transformer is located in front of the longitudinal side of the building facing the turbine building.

The turbine building is located adjacent to the reactor building with the turbine axis pointing in the direction of the reactor building. This gives maximum protection of the reactor building should the highly unlikely event of a turbine rotor burst occurs.

The building is of a two bay design. The main bay houses the turbine generator set and the feedwater heating equipment. The lower ancillary bay houses the feedwater tank, deaerators and feedwater pumps and other equipment associated with the water/steam cycle. All these compartments are free of radioactive media.

The main steam lines coming from the reactor building enter the turbine building along the shortest route leading to the area of the high pressure casing of the turbine, where the main steam flows through the steam strainers into the high pressure turbine, Vertical moisture separators are installed on both sides of the high pressure casing.

The basement of the turbine building is used mainly to accommodate pipes and cables. The heat exchangers for the low pressure feed heater drains, the closed circuit cooling water system and the associated pumps are also installed in the basement.

The generator busbars are routed from the generator to the generator transformers installed against the wall outside the turbine building, and to the high voltage station service transformers.

The circulating water pipes enter and leave the turbine building on the same side.

The emergency power and chilled water supply building has two service floors. The building is further subdivided into four equal sections of similar construction which house redundant systems and equipment.

The diesel fuel storage tanks, pumps, secured component cooling heat exchangers, air recirculation system and the cable and pipe spreading rooms are installed on the lower floor.

The emergency power generators with their switchgear and the water-chilling units are installed on the upper floor. The diesel fuel day tanks and the start-up air supply system for each of the emergency power generators are installed on a gallery structure.

IV.3. CNA II MAIN SYSTEMS

IV.3.1. Reactor

The reactor (*Figure IV-4*) is of the pressure vessel type, natural or slightly enriched uranium fuelled and heavy water cooled and moderated. The total thermal power is 2160 MW.

The reactor core is approximately cylindrical in shape and consists of 451 natural uranium fuel assemblies located in the same number of coolant channels. Each fuel assembly consists of 37 fuel rods arranged in three concentric circles, the rod supporting plate, the spacers for lining up the fuel rods, and the linkage with a coupling for connection to the filler body. Each fuel rod consists of a stack of uranium dioxide pellets enclosed by a thin walled zircaloy 4 canning tube, which is both gas and pressure tight. Each fuel assembly, together with the filler body and the closure plug, forms the fuel bundle column. The coolant channels are arranged vertically in a triangular lattice within the moderator tank. The fuel bundle columns can be removed from the coolant channels during reactor operation by the refueling machine. The filler bodies serve to reduce the volume of the coolant in the reactor coolant system.

The heat generated in the fuel assemblies is transferred to the reactor coolant, which flows through the coolant channels and transports the heat to the steam generators.

The coolant channels are surrounded by the moderator, which is enclosed in the moderator tank. For reactivity reasons, the moderator is maintained at a lower temperature than the reactor coolant. This is accomplished by the moderator system, which extracts the moderator from the core, cools it down in the moderator coolers, and feeds it back into the core. The heat removed from the moderator is used for pre-heating the feed-water. This is one of the reasons for the high net efficiency (approx. 32%) of the NPP.

The reactor coolant system and the moderator system are connected by the pressure equalization openings of the moderator tank closure head. Therefore, the pressure differences in the core are comparatively small, which results in thin walls for the reactor pressure vessel internals. This allows a very high burn-up to be attained. Furthermore, the connection between the reactor coolant system and the moderator system permits the use of common auxiliary systems to maintain the necessary water quality. The number of auxiliary systems can therefore be reduced to a minimum.

For control of the reactivity, and thus of the power output of the reactor, various methods are applied. The reactor contains nine "blacks" (absorbers made of hafnium) and nine "grey" (steel) control

elements arranged in 3 groups. The control elements are used to control the reactivity and the power distribution, to compensate the build-up of xenon poisoning following a reactor power reduction, to provide damping of azimuthal xenon oscillations, and to shut down the reactor. The reactivity value of all control elements is sufficient to shut the reactor.

In addition to the control elements, reactivity control is provided by the boric acid dosing system. The injection or extraction of boric acid serves to compensate slow reactivity changes due to the burnup during the first period of operation and to maintain the reactor in a safe subcritical condition at zero power. Extraction of the boric acid is performed by anion exchangers.

Additionally, a boron injection system, as a second independent shutdown system is provided, which injects boric acid into the moderator.

The reactivity can, in addition to these reactivity control systems, also be controlled by varying the moderator temperature within a certain range, which is advantageous for some operating modes.

The reactor pressure vessel constitutes the pressure boundary of the reactor core and encloses the core components and the reactor pressure vessel internals. The RPV consists of the lower part, the closure head and the studs and nuts which connect both sections. The connection is made leak-tight by means of a welded lip seal.

The lower part of the RPV consists of the hemispherical bottom section, two shell courses and a shell flange which carries the coolant inlet and outlet nozzles and the support pads located between them. The reactor coolant inlet and outlet nozzles are arranged on one plane; there are no penetrations or pipe connections below this plane. The reactor core is housed below the plane of the inlet and outlet nozzles.

The closure head consists of a flange and a dome plate connected by a circumferential weld. The closure head dome carries the nozzles for coolant channels, moderator pipes, and control element drives and for in-core instrumentation. The nozzles are screwed into holes in the closure head dome and sealed by an overlay weld.

Most of the RPV internals form the structure of the reactor core. The moderator tank accommodates all core components, separates the moderator from the coolant and, in conjunction with the reactor pressure vessel, forms the annulus for the in-flowing coolant. The bottom of the moderator tank serves as the lower fixing level for the coolant channels and the control element guide tubes. The moderator tank shell serves as thermal shielding.

The moderator tank closure head 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 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 required. 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.

IV.3.2. Reactor coolant system and moderator system

The reactor coolant system ([Figure IV-5](#)) removes the heat generated in the reactor core and transfers it via the steam generators to the turbine generator plant.

The system is structured similar to that of a pressurized light water reactor and consists of two identical loops, each comprising a steam generator, a reactor coolant pump and the interconnecting piping, as well as one common pressurizer and pressurizer relief system.

The heat is carried by the reactor coolant, which flows from the reactor pressure vessel to the steam generators, where it is cooled down and then pumped back to the reactor pressure vessel by the reactor coolant pumps.

The pressurizer system is connected to one reactor coolant loop and basically comprises the pressurizer with the electric heaters, the surpline, the spray lines with valves, and the safety valves.

The function of the pressurizer system is to maintain the appropriate pressure in the reactor coolant system in order to prevent boiling of the coolant under all operating conditions (principle of the pressurized water reactor), and to avoid or limit the pressure variations caused by volume fluctuations during load changes. The pressurizer is partly filled with saturated water and partly with steam. If the pressure drops, water is evaporated by switching on the electric heaters, raising the pressure to its set point. In the event of a pressure rise, steam is condensed by spraying water into the steam space.

Besides pressure control by sprays in the pressurizer, protection against overpressure in the reactor coolant system is provided in accordance with international codes for pressure vessels and systems. Protection is afforded by independent, self-actuating safety valves.

When the safety valves open, the steam discharged from the pressurizer is directed into the pressurizer relief system, where it is condensed to water.

The moderator system consists of four identical loops operating in parallel. Each loop comprises a moderator cooler, a moderator pump, and the interconnecting piping with valves.

The moderator system performs various functions depending on the operating mode of the reactor.

During normal operation, the moderator system maintains the moderator at a lower temperature than that of the reactor coolant. The moderator leaves the top of the moderator tank, flows to the moderator pumps, is pumped to the moderator coolers and flows back to the bottom of the moderator tank. The heat transferred in the moderator coolers is used for preheating the feedwater.

For residual heat removal, the moderator system is switched over to the residual heat removal position by means of the moderator valves. Under this mode of operation, the moderator is extracted from bottom of the moderator tank by the moderator pumps and fed into the cold legs of the reactor coolant loops, and also directly into the reactor coolant inlet annulus of the reactor pressure vessel, via the moderator coolers. The moderator system forms the first link of the residual heat removal chain. The residual heat is transferred from the moderator system to the residual heat removal system and then to the service cooling water system.

During emergency core cooling, the moderator serves as a high pressure core reflooding and cooling system. The emergency core cooling position is similar to that of the residual heat removal, but additionally, water is injected into the hot legs of the reactor coolant loops and into the upper plenum of the reactor pressure vessel. The residual heat removal chain connected to the moderator coolers during emergency core cooling is the same as during residual heat removal.

The steam generators transfer heat from the reactor coolant on the primary side to the feedwater/steam cycle on the secondary side. The transferred heat raises the feedwater temperature and generates the saturated steam which drives the turbine generator unit. The steam generator constitutes the barrier between the radioactive reactor coolant and the non-radioactive feedwater/steam cycle, preventing the carry over of radioactive matter.

The steam generator is a vertical U-tube heat exchanger with natural circulation of the feedwater on the secondary side. The primary side of the steam generator consists of the channel head (primary plenum) and of the heating tube bundle. On the secondary side, the feedwater enters through a nozzle located in the steam dome and is distributed by a feedwater ring manifold in the annulus, (down-comer) formed by the secondary side shell and the tube bundle wrapper. The feedwater flows downwards to the tube sheet and enters the tube bundle region (riser). The feedwater is heated and partly evaporated around the U-tubes in this region. The generated steam-water mixture leaves the riser region and flows through cyclones and steam driers arranged in the steam dome. The dried

steam is discharged through the main steam outlet nozzle, the separated water flows downwards into the down-comer where it is mixed with the incoming feed-water.

All systems of the residual heat removal chain are of a consistent “four loop” design. The residual heat removal system acts as a barrier between the active moderator and the service cooling water and prevents the escape of radioactivity into the service cooling water in the event of leakages in the moderator coolers.

IV.3.3. Refuelling system

The natural (and slightly enriched) uranium reactor makes it possible and desirable, with a view to obtaining a high burnup, to shuffle and replace the fuel assemblies during power plant operation. The refueling procedure is carried out by a single refuelling machine. The fuel assembly transport system is located in the reactor building and in the fuel pool building. The main items of the fuel transport system are: refuelling machine, tilter with supporting structure, fuel transfer tube, fuel pool, and the corresponding auxiliary systems and maintenance installations. The refueling procedure is fully automated and monitored from the control room.

The refueling machine is moved from a maintenance position in the refueling machine maintenance room, by remote control, to a previously selected coolant channel position in the reactor vessel. in which the machine is centered. The seat-on seal is pressed on to the coolant channel closure head by the dead weight of the refueling machine to form a water-tight seal between the machine and the coolant channel. A pressure equalization takes place between the refueling machine and the reactor before opening the isolation valve of the refueling machine and opening the coolant channel closure. Following this, the fuel bundle column is withdrawn into an empty position in the refueling machine magazine. The magazine is then rotated in such a way that a fuel bundle column with a partially burnt up fuel assembly or with a new fuel assembly is positioned above the open coolant channel. This fuel bundle column is lowered into the coolant channel position and the coolant channel closure is locked again. After closing the isolation valve of the refueling machine a check for leak tight closure is performed. Then the refueling machine is removed from the reactor pressure vessel and is positioned above the vertically arranged titter. The titter has the following functions in the sequence indicated:

- Take-over of the fuel bundle column with the spent fuel assembly.
- Removal of the decay heat by cooling with D_2O .
- Drying and cooling the spent fuel assembly with gas.
- Flooding and cooling of the titter with H_2O .
- Tilting to the horizontal position and connecting with the fuel transfer tube.
- Transfer of the fuel assembly into the fuel transfer tube.

When a new fuel bundle column is transported from the fuel pool building into the titter via the transfer tube, and later from there into the refueling machine, the process of cooling and change of cooling medium takes place in the reverse order.

The task of the fuel transfer tube is to establish a connection between the tilter within the reactor building and the tilting device in the fuel pool building, while both components are in the horizontal position.

The tilting device takes the fuel bundle column from the fuel transfer tube and swivels it from the horizontal into the vertical position.

A silver-clad seal ring with good material flow properties is used as sealing material. In this established design, the coolant channel seals are almost perfectly tight.

IV.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.

IV.3.5. Instrumentation and control systems

The instrumentation and control equipment includes the measurement, control, protection and monitoring system.

The control room is, through the automation and plant interface equipment, connected to the drives and signal transmitters in the plant. Plant conditions and operational transients are transmitted via analog and binary signal transmitters to both the operator in the control room and to the automatic equipment. The command signals to the drives in the plant are transmitted via the control interface as required for maintaining proper operation.

Measured-value and status signals from the entire plant are continuously monitored by means of a process computer. The computer provides the data for trend logging and fault analysis and transmits the information to the operating personnel in the control room via data display terminals and teleprinters.

Automatic functional group controls are provided to minimize the operating errors and to obtain a higher degree of automation.

IV.3.5.1. Main control room

The main control room of the nuclear power plant contains the operating and information equipment for the control and monitoring of the plant systems. This means that manual control, setpoint adjustment and monitoring of the reactor, important reactor auxiliaries, the feedwater/steam cycle, the turbine, the generator and the auxiliary power equipment are controlled from the main control room.

The main control room is situated on the top floor of the switchgear building above the electronic equipment rooms. For security reasons, it shall only be entered through monitored entrances.

IV.3.6. Electric power system

The CNA II has two physically independent grid connections. One of them is the 500 kV grid and the other is the 132 kV grid.

There is additionally, auxiliary power supply from the generator in case of grid disturbance after load rejection. Only in the case of a common outage of all three power supply possibilities, the emergency power system with the diesel generators will be required. Definite loads – mainly of the control and instrumentation system – are power supplied by rectifiers and converters or by means of batteries with direct current.

The generator feeds into the 500 kV network via one generator transformer and supplies the station auxiliary service requirements by means of two auxiliary transformers.

The four secondary windings of the auxiliary transformers or the two secondary windings of the offsite system transformer feed into four separate medium high voltage bus section (each two 6.6 kV and 13.2 kV), to which the large auxiliary loads and the transformers for the low voltage switchgears are connected.

If the station service power system fails, certain equipment (pumps, etc.) are needed to remove residual heat and to run the plant into safe conditions. This equipment must remain in operation or has to be put in operation and must therefore be supplied with emergency power. Here a distinction must be made between two groups with regard to safety related requirements: loads allowing a voltage interruption while the diesel run up and loads which must remain in operation without interruption or which must be put into operation immediately should the normal station service system fail.

The diesel emergency power system is like the other redundant safety equipment, divided into for separate trains. Under normal operating conditions, the auxiliary switchgears of the normal power system

feed the emergency power system. To avoid loss of power in case the auxiliary power system fails, each of the four trains in the emergency power system is equipped with a quick-starting diesel set.

The CNA II electric system may be divided into two main subsystems: the offsite power system and the onsite power system.

The offsite power system is constituted by the 500 kV transmission line, which is linked to the substations Rosario Oeste (113 km), Colonia Elía (160 km) and Ezeiza (67 km), and the 132 kV transmission line, source that is connected to the Zárate substation (23 km) and, in addition, to the 220 kV switchyard at the power plant Atucha via a 150 MVA coupling transformer.

With the generator load-breaker in the “off” position the station service power for “start up” and “shut down” of the power plant can be drawn from the 500 kV grid. Upon simultaneous failure of the main grid and the turbine generator set the 132 kV grid provides power for shut down operation of the plant down to the “hot-subcritical” condition.

The onsite power system consists of two subsystems: the auxiliary power system (“normal system”) and the emergency power system.

The auxiliary power system provides power for the loads of the nuclear power plant which are necessary during normal operation, start-up and shut-down operations. It is subdivided into four trains which are supplied by the 13.2 kV and 6.6 kV windings of the two unit auxiliary transformers. The transformers are fed in via single phase totally enclosed leads either from the min generator or from the grid via the external generator transformer. For shut-down operation or after loss of the normal power supply grid and the generator, it may be fed by the offsite power supply. The offsite power supply is available via automatic changeover.

The A.C. emergency power system provides the power required for safe shut-down of the reactor to maintain it in the shut-down condition, for removal of residual heat and to prevent release of radioactivity during normal operation and accident conditions resulting from system faults, and for some loads important for plant availability. It is subdivided, according to the safety system to be power supplied, into four redundant independent trains, each capable of supplying 50% of the power required to perform the safety function.

Normally the A.C. emergency power supply system is connected with the two 6.6 kV buses of the plant auxiliary power system by means of two circuit breakers, connected in series, for each train. Therefore, it can be fed via the plant auxiliary power system or the offsite system. At loss of the plant auxiliary power it is fed by the diesel generator emergency power system supply.

The stand by diesel generator emergency power system is provided for safety related loads. Each diesel generator set is assigned to one train, each with 50% capacity. Each emergency diesel generator is constituted by different main and auxiliary subsystems, such as the compressed air, the fuel supply equipment, the lubrication system, etc.

In the event of the power plant and the high voltage system failing at the same time, an emergency power supply can be obtained from diesel sets to permit proper shutdown of the plant. On failure of the auxiliary voltage, a period of 20 to 30 seconds elapses before the emergency diesel sets can take load. However, certain loads, such as the reactor protection system, measuring systems, and other protective systems must remain operative at all times. These are therefore fed directly from 220 V or 24 V D.C. batteries or from static converters fed from the 220 V breakers.

IV.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:
 - o Conservative and careful design.
 - o Stringent quality assurance and control.

- o 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:
 - o Inherently safe operational characteristics.
 - o Alarm annunciation.
 - o 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 IV-6*).

The emergency electric power system is not a safety system but a safety related one due the nature of the loads it feeds. It was described in section IV.3.6.

The general plant layout features a compact, controlled area with short pipe and cable connections, physical separation of redundant cable and piping trains, minimum exposure of the operating personnel during maintenance and repair work, availability of sufficient shielding and radiation protection measures.

IV.4. TECHNICAL DATA

Some of the main technical data are detailed in what follows:

Overall Plant Data	
Reactor type	Pressurised heavy water (PHWR)
Bulk nominal electric power	745 MW
Net nominal electric power	692 MW
Net efficiency	32%
Number of steam generators	2
Number of reactor coolant pumps	2
Number of moderator coolers and moderator pumps	4

Reactor Core Data	
Material of fuel	Natural uranium
Total thermal power	2160 MW
Shape of fuel assembly	37 – rod cluster
Number of coolant channels or fuel assemblies	451
Cladding material	Zircaloy 4
Active core length	5300 mm
Refuelling	On load
Fuel burnup at equilibrium	7500 MWd/MgU
Number/material of control rod elements	9/hafnium and 9/steel

Thermal and Hydraulic Data	
Number of coolant circuits	2
Number of moderator circuits	4
Coolant and moderator	Heavy water
Pressure at reactor vessel outlet	115 bar
Coolant temperature at reactor pressure vessel outlet	312° C
Coolant temperature rise though the core	34° C
Moderator temperature normal/maximum	170°/220° C
Total coolant circulation flow	37080 Ton/h
Total moderator circulation flow	3200 Ton/h
Steam pressure at steam generator outlet	56 bar
Steam temperature	271° C
Total steam flow	3445 Ton/h

Steel Containment Data	
Diameter	56 m
Wall thickness	30 mm

Design pressure	5.3 bar
Design leak rate	0.25 vol %/day

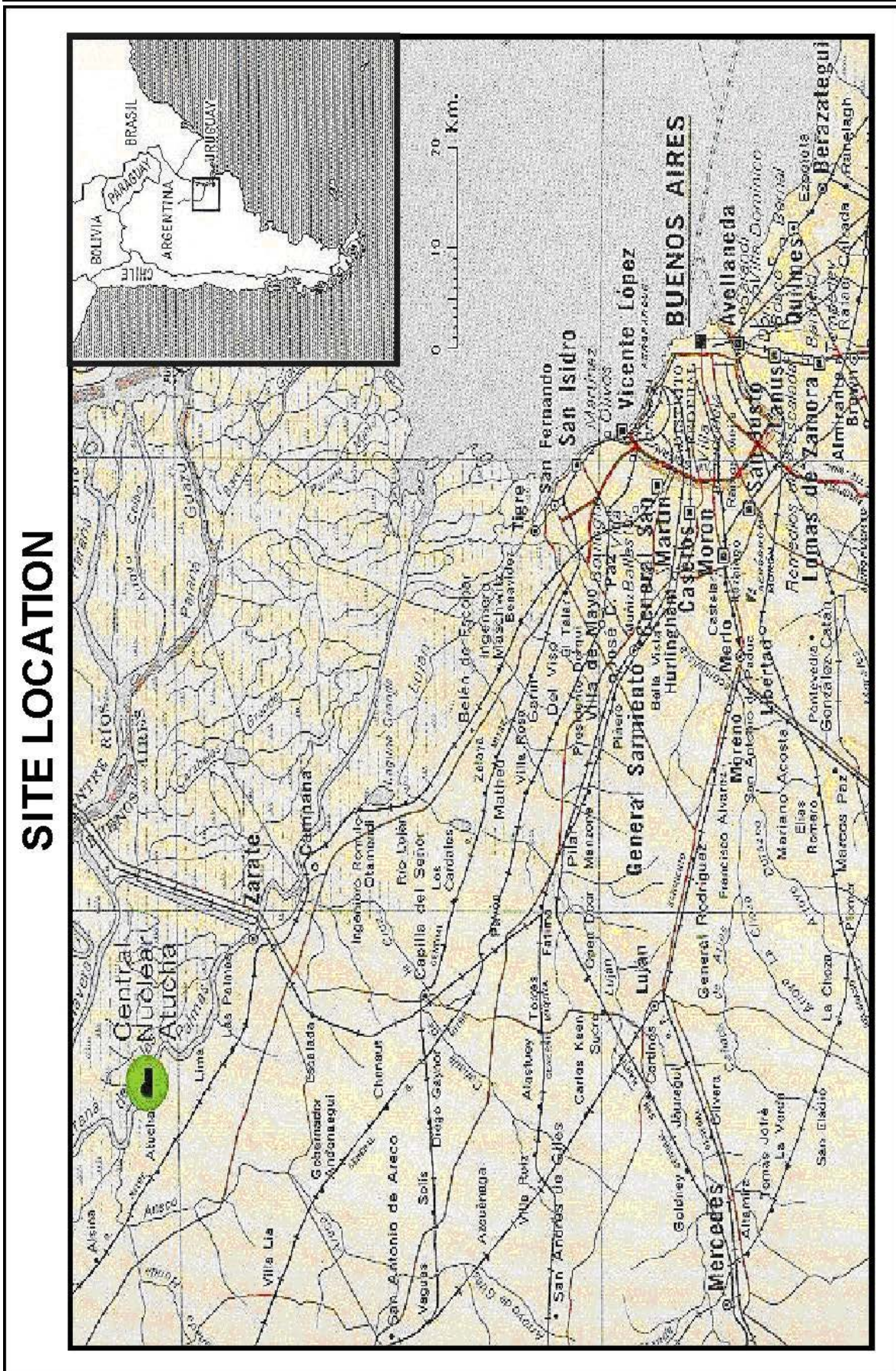


Figure IV-1 - Atucha II Nuclear Power Plant - Geographic Location

CNA II SIMPLIFIED FLOW DIAGRAM

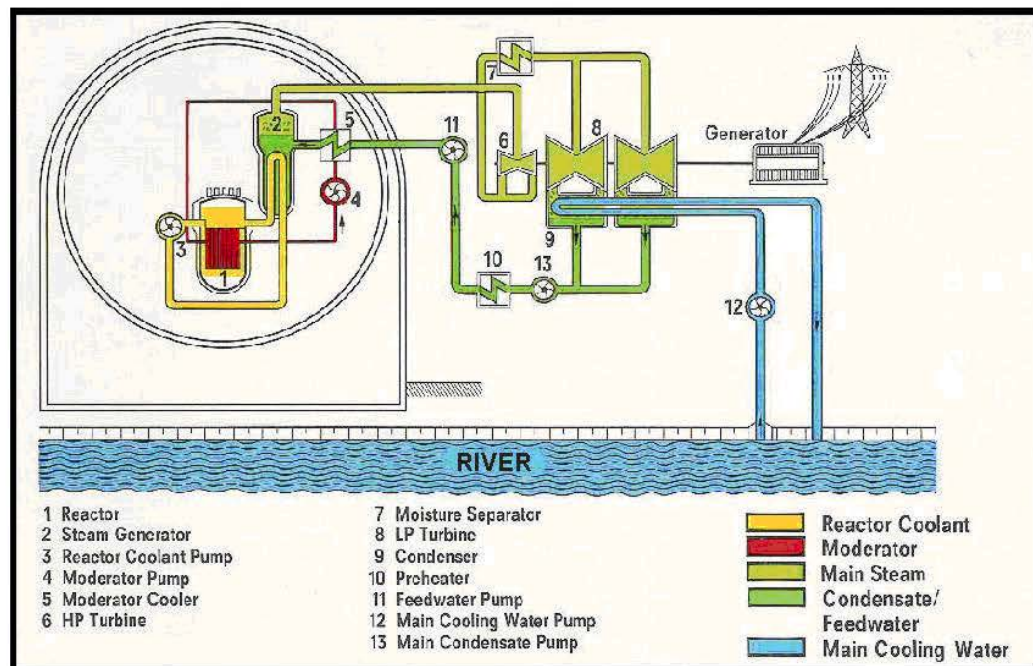


Figure IV-2 - Atucha II Nuclear Power Plant - Main Systems

SITE PLAN

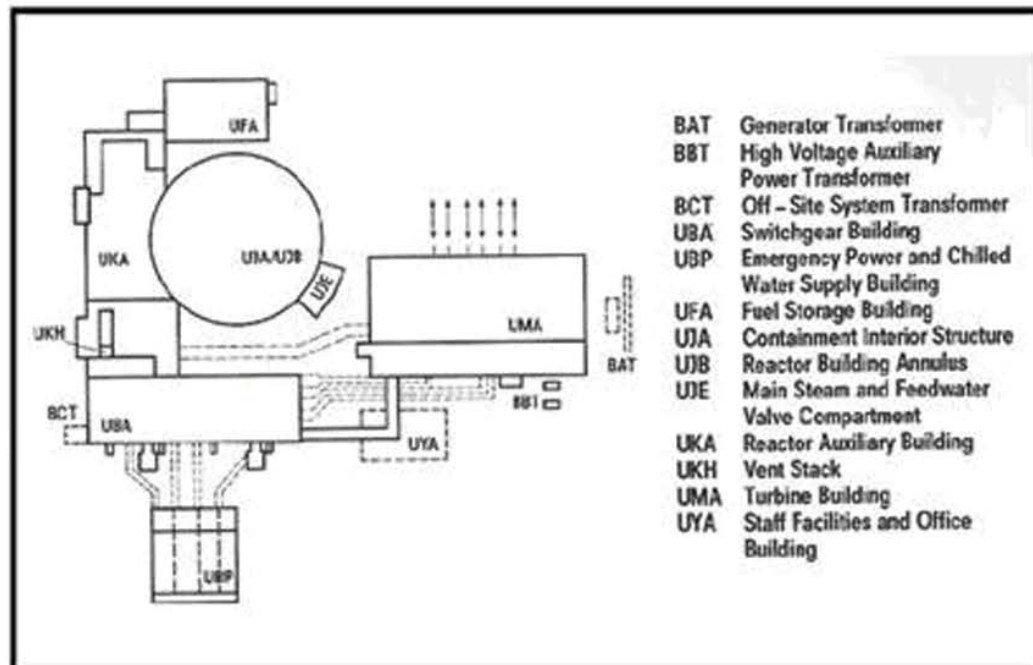


Figure IV-3 - Atucha II Nuclear Power Plant - Main Building and Structures

REACTOR PRESSURE VESSEL - INTERNALS

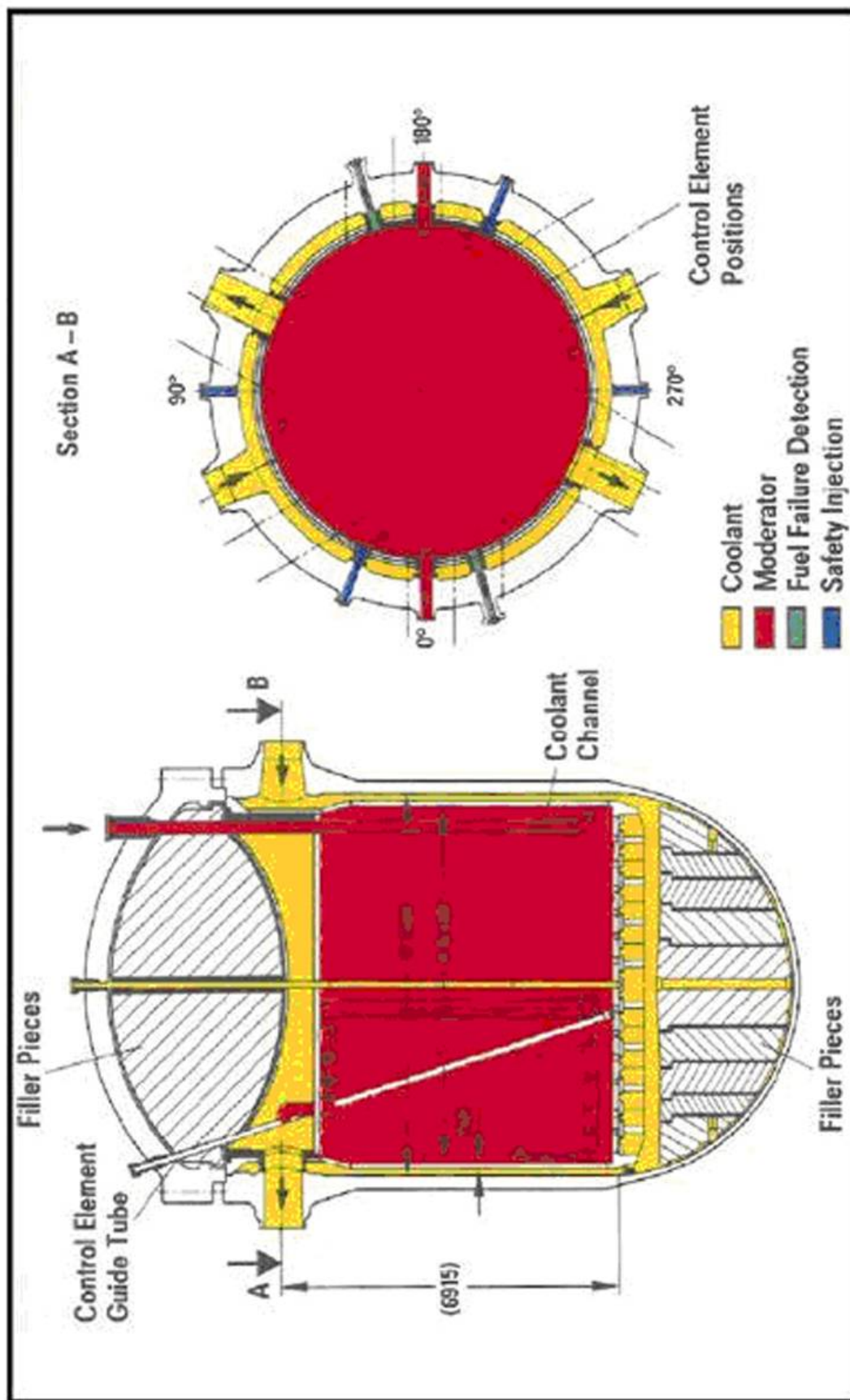


Figure IV-4 - Atucha II Nuclear Power Plant - Reactor Pressure Vessel with Internals

PRIMARY SYSTEMS - NORMAL OPERATION

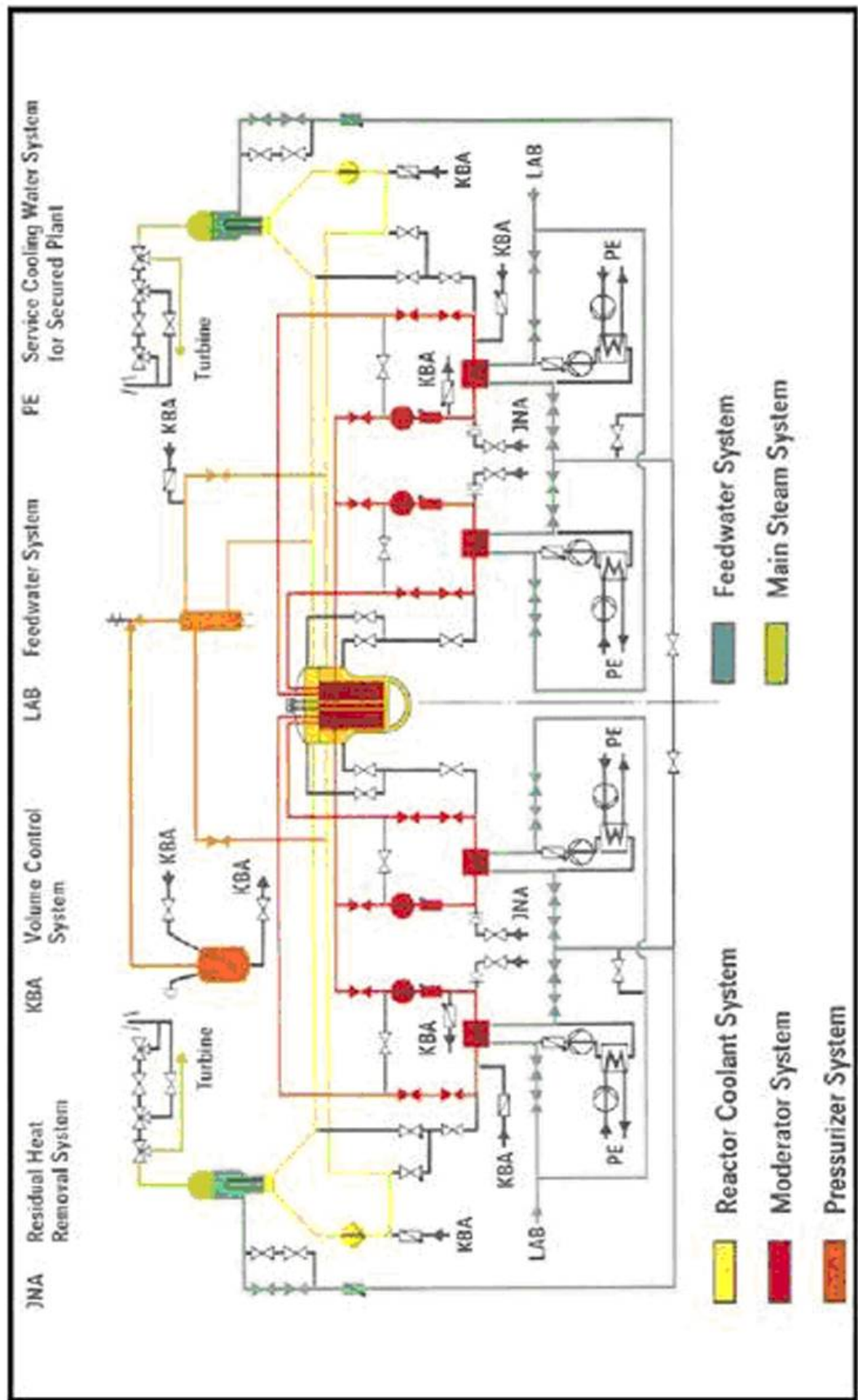


Figure IV-5 - Atucha II Nuclear Power Plant - Primary Systems - Normal Operation

REACTOR BUILDING

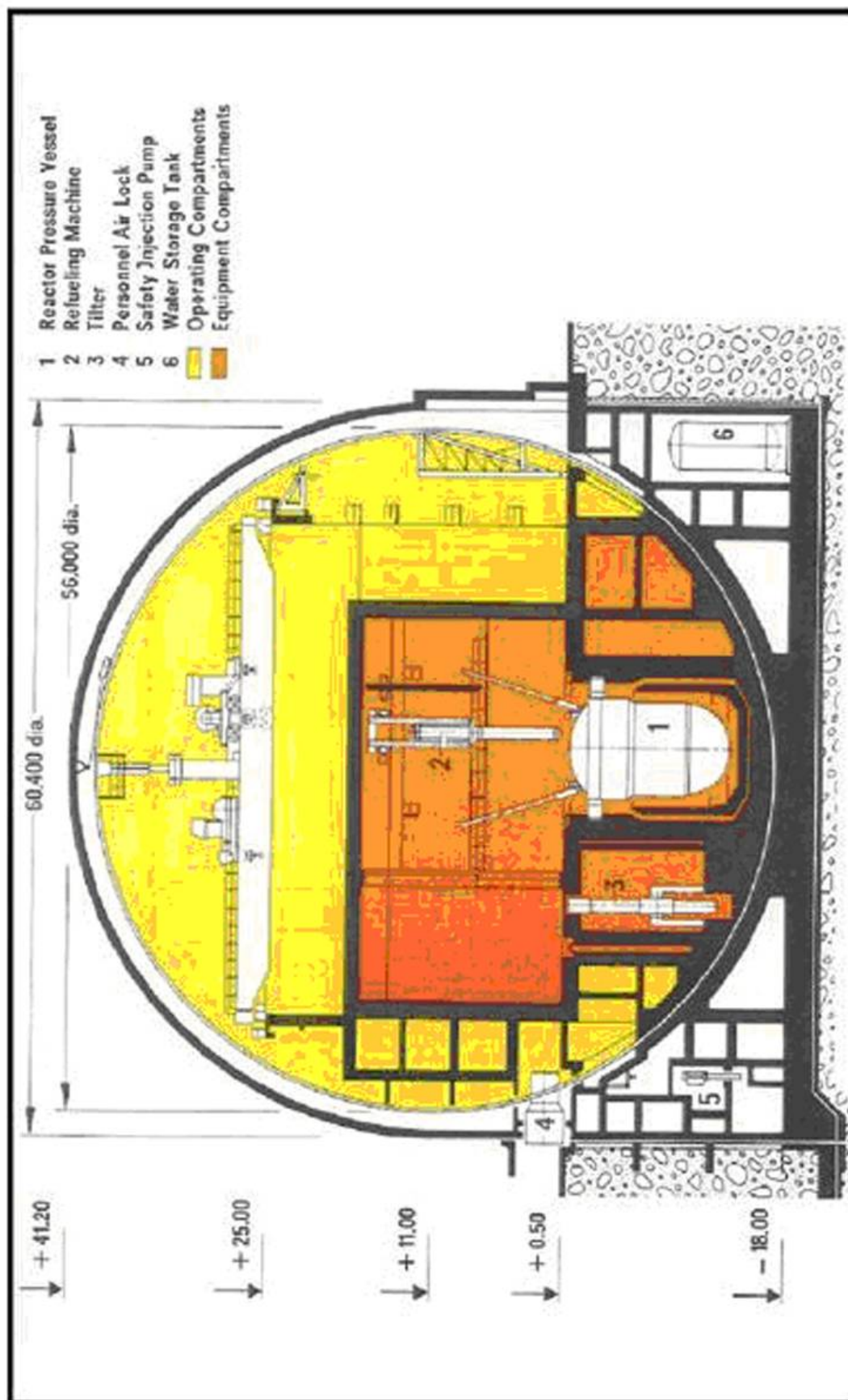


Figure IV - 6 - Atucha II Nuclear Power Plant - Reactor Building

ANNEX V

PRINCIPAL TECHNICAL CHARACTERISTICS OF CAREM REACTOR PROTOTYPE

V.1. INTRODUCTION

CAREM reactor prototype (CAREM-25) has an indirect cycle reactor with some distinctive and characteristic features that greatly simplify the design, and also contributes to a higher safety level. Some of the high level design characteristics of the plant are: integrated primary cooling system, natural circulation, self-pressurised primary system and safety systems relying on passive features.

V.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 V-1**).

The flow rate in the reactor primary systems is achieved by natural circulation. Figure V-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.

V.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 V-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 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 390 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).

V.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 V-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.

V.5. REACTOR AUXILIARY SYSTEMS

Figure V-4 shows a diagram of the main reactor auxiliary systems chemical / volume control system.

V.5.1. Chemical / volume control system

This system maintains a high degree of water purity within the RPV and allows controlling the water level while volume changes are produced by the operating conditions. The water removed from the RPV is cooled in a letdown heat-exchanger, undergoes a stage of pressure reduction, is treated in filters, resin beds, de-gasifier columns and returned to the RPV by the charging pumps through a regenerative heat exchanger. A control volume tank provides a volume reservoir that allows to contain all the water surplus of the RPV from the operation in a solid way at 45 °C until full power operation.

V.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.

V.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*.

V.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*.

V.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.*

V.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.

V.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.

V.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.

V.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.

V.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.

V.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 admitted.
- 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.

V.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 V-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

V.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 Systems equipment. In CAREM-25 these six levels are described below from 11.1.1 up to 11.1.6:

V.11.1.1. Level +15.20: *It is* the Reactor Hall where tasks related with the refuelling will be performed.

V.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.

V.11.1.3. Level +5.20: Valves rooms for the filters and resins beds, *electrical swichboards and Standby Gas Treatment System*.

V.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*.

V.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*.

V.11.1.6. Level -10.10: Pumps, Filters and Heat Exchangers of Safety-Related systems are housed in this area.

V.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 Drywell 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 and the SG's inspection robot.* 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.

V.11.2. Turbine module

It houses the turbo-generator group, auxiliary services such as de-mineralised water production system, chilled water, service steam, condensate polishing, and electric switchboards.

Close to the Reactor Module are the redundant diesel-generators with switchboards.

V.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.

V.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.

V.13. SAFETY SYSTEMS AND FEATURES

The safety systems are duplicated to fulfil the redundancy criteria (**Figure V-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 V-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 two tanks located in the upper part of the containment. Each of them 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 tank. When the system is triggered, the valves open automatically and the borated water drains into the primary system by gravity. The discharge of a single tank 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-covering 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 V.I.**

Table V.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 <i>Operated Relief Valve</i>
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 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 steam 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 not 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.

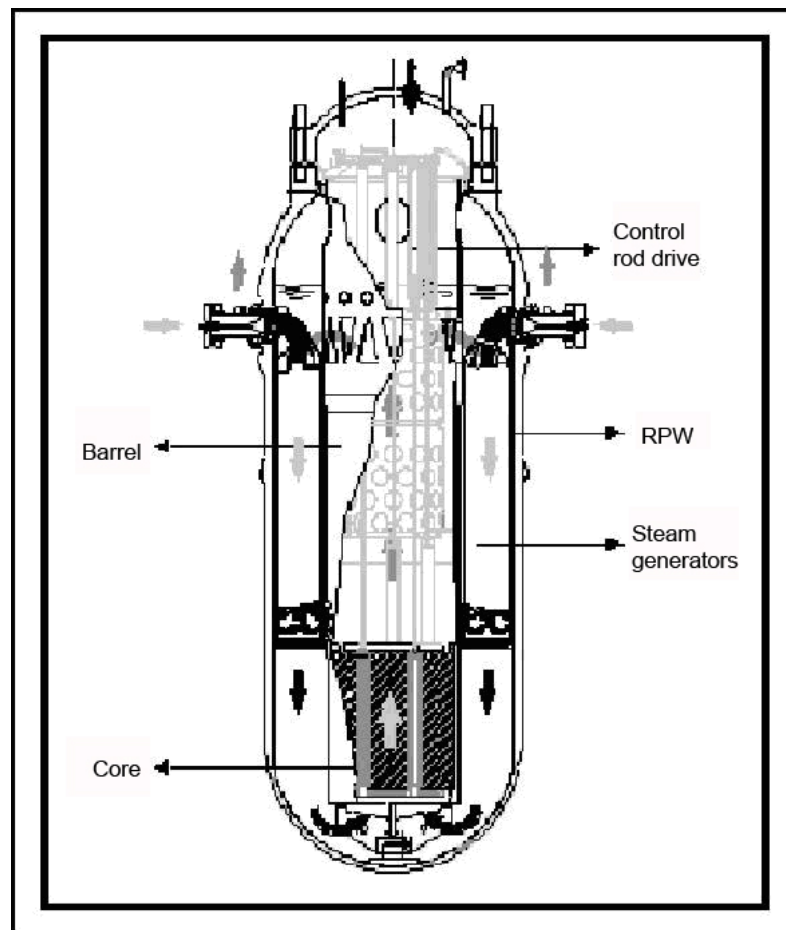


Figure V.1 - Reactor Pressure Vessel

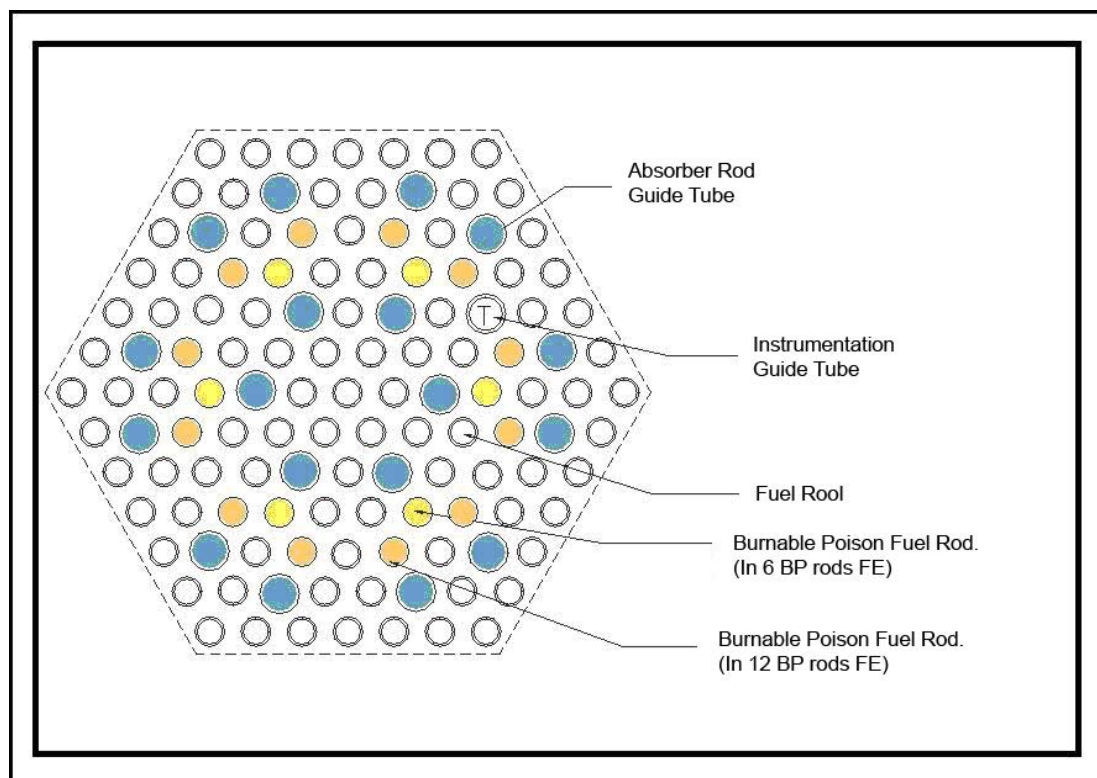


Figure V.2 - Fuel Assembly Diagram. Fuel rods, guide thimbles and instrumentation thimble distribution

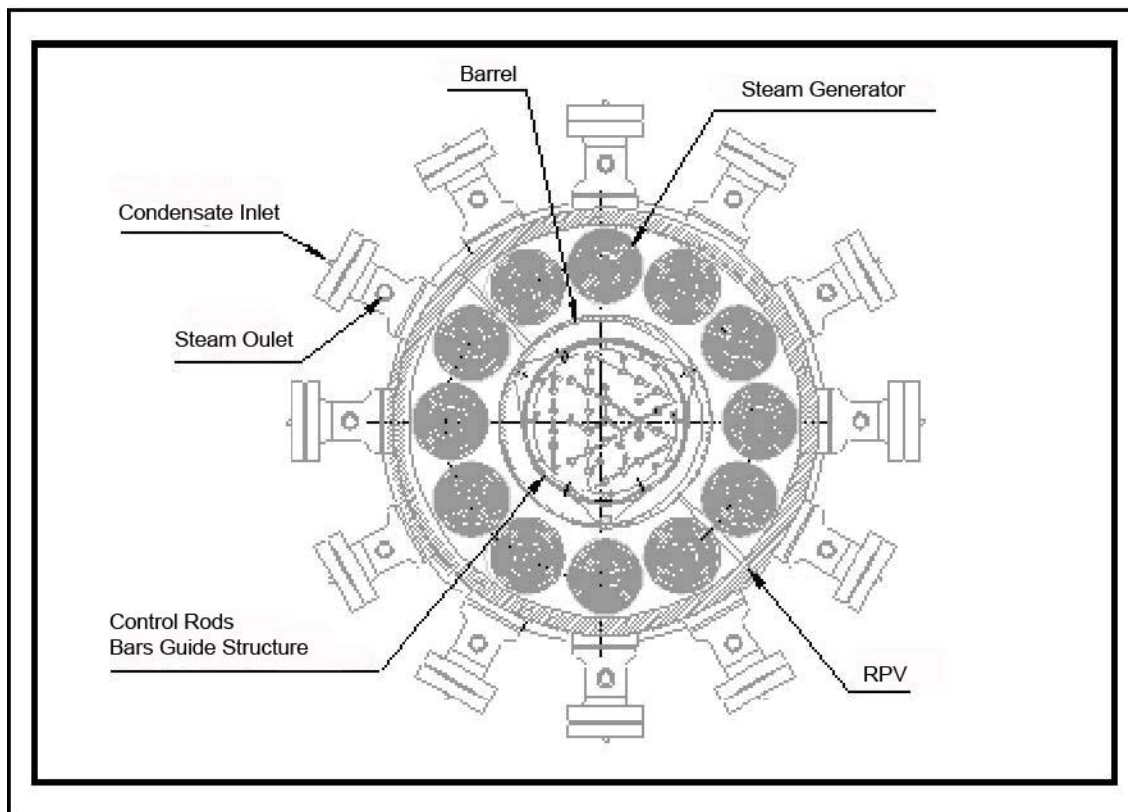


Figure V.3 - Steam Generation Layout

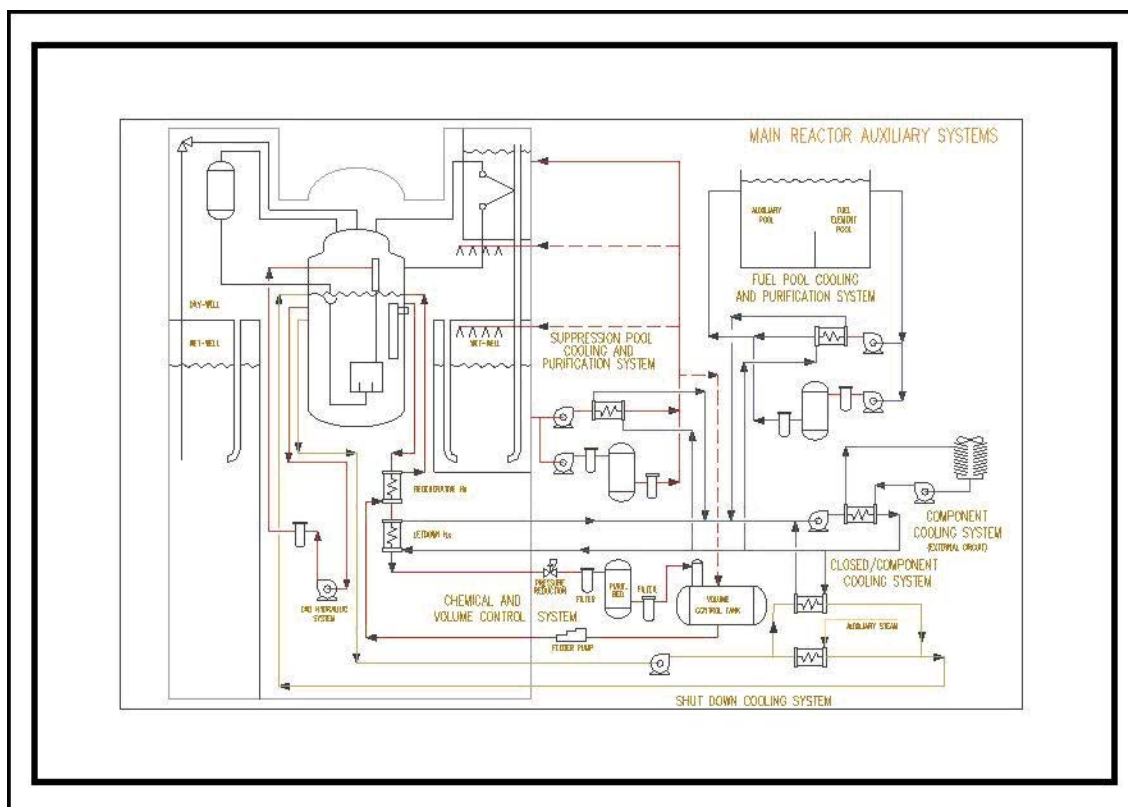


Figure V.4 - Auxiliary Systems

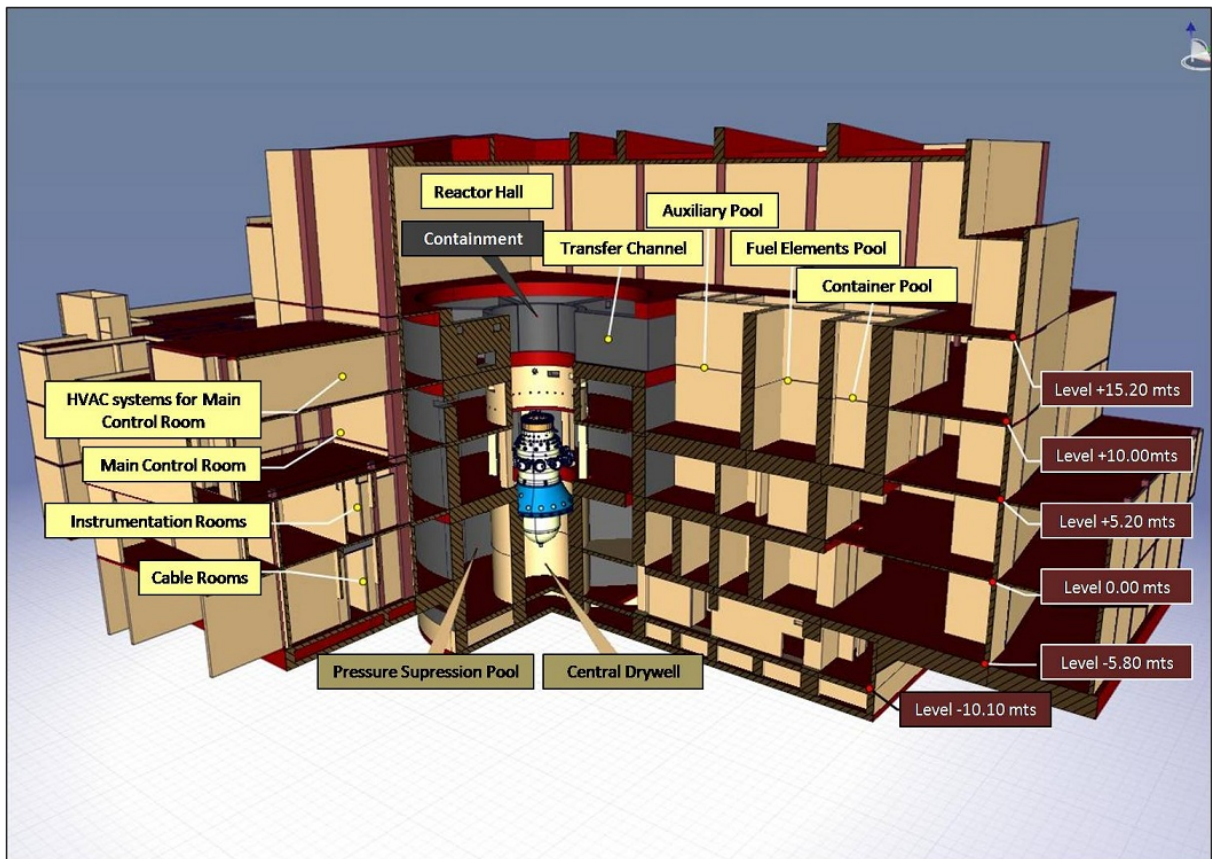


Figure V.5 - Plant Layout

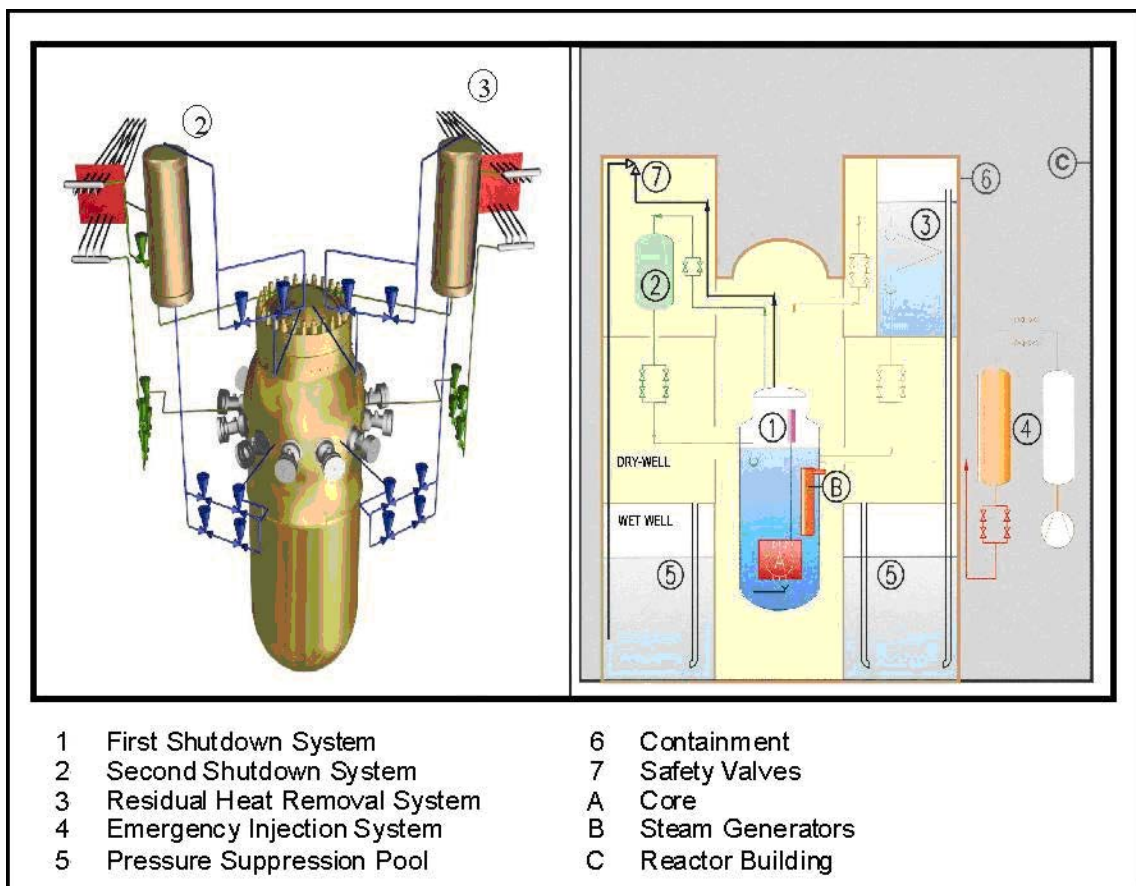


Figure V.6 - Containment and Safety Systems

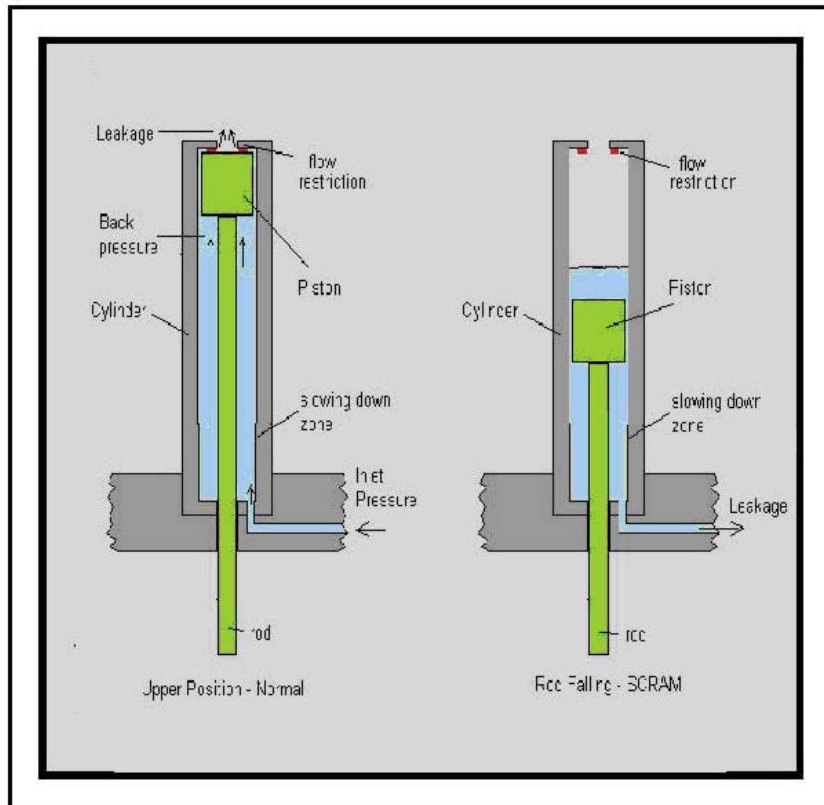


Figure V.7 - Simplified Operation Diagram of a Hydraulic Control Rod Drive (Fast Shutdown System)

ANNEX VI

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 and CNE during the period 2010 – 2012, are listed and how the licensees and the ARN consequently and accordingly acted, are shown.

VI.1. EXAMPLES OF LESSON LEARNED AND CORRECTIVE ACTIONS RESULTING FROM INTERNATIONAL EVENTS AND OPERATING EXPERIENCE

<i>EVENT - OPERATING EXPERIENCE</i>	<i>EXAMPLES OF LESSONS LEARNED - CORRECTIVE ACTIONS</i>
<i>High-voltage bushing failure results in main transformer fault and unplanned outage (Indian Point Unit 3, PWR, April 2007)</i>	<i>CNA I: An on line monitoring system with bushing sensor is planned to be bought.</i>
<i>Underground cable ground fault causes forced shutdown (Point Beach Unit 1, PWR, USA, January 2008)</i>	<i>CNA I: Visual inspections of MT cables routes were established.</i>
<i>SDS 1 Channel D tripped on regional over-power trip loop 8 due to loose connections (Cernavoda 2, PHWR, Romania, November 2009)</i>	<i>CNE: Connectors (LEMO and BNC) state was verified in panel 68220 PL111 room S-325.</i>
<i>Damaged life buoys (Gentilly 2, PHWR, Canada, July 2010)</i>	<i>CNA I: A periodic inspection protocol was made to inspect every water rescue element in the plant.</i>
<i>Risk of water hammer when ECC pump stops and restarts during low-pressure of emergency core cooling (Gentilly 2, PHWR, Canada, October 2010)</i>	<i>CNE: Change recommendations on emergency core cooling during low-pressure phase are under detailed analysis process.</i>
<i>Fire hydrant main leak (Hunterston B, AGR, England, December 2010)</i>	<i>CNA I: Procedural modification to inspect buried piping in zones where heavy cargo movement is to be done.</i>
<i>Run-up to Power with Heat Transport System pH outside of limits following the 2005 planned outage (Gentilly 2, PHWR, Canada)</i>	<i>CNE: An addendum was append to Operations Manual to avoid a significant pH drop during an outage.</i>
<i>Increase in loss of decay heat removal events (Fort Calhoun1, PWR, USA)</i>	<i>CNE: Lessons learned were disseminated to Operations and Nuclear Safety staff.</i>
<i>Water leak in pump motor cooler leads to unit shutdown (Bruce 3, PHWR, Canada, February 2011)</i>	<i>CNE: Organization chart is under review in order to assign resources to work on obsolescence and equipment ageing.</i>
<i>Unavailability of Heat Transport Feed Circuit Pump due to Lack of Lubrication during Pump Shutdown (Bruce A Unit 4, PHWR, March 2011)</i>	<i>CNA II: Lessons learned were disseminated to Operations, Engineering and Commissioning Mechanical staff.</i>

EVENT - OPERATING EXPERIENCE	EXAMPLES OF LESSONS LEARNED - CORRECTIVE ACTIONS
<i>Damaged lighting conduit and cable (Darlington 1, PHWR, Canada, April 2011)</i>	<i>CNE: A buried cable detection instrument was bought and made operative. CNA I: Lessons learned were disseminated to Engineering and Industrial Safety staff. CNA II: Lessons learned were disseminated to Operations, Engineering and Maintenance staff.</i>
<i>Moderator purification strainer corroded and fell apart (Pickering 5, PHWR, Canada, April 2011)</i>	<i>CNA I: A verification of filter components replacement stock will be made before maintenance work order tasks begin.</i>
<i>Testing of digital integrated level control system results in reactor scram (Susquehanna 1, BWR, USA, April 2011)</i>	<i>CNA I: Lessons learned were disseminated to I&C maintenance shop staff.</i>
<i>Condensate storage tank level controller failed in unsafe mode (Bruce 7, PHWR, Canada, June 2011)</i>	<i>CNA I: Lessons learned were disseminated to I&C maintenance shop staff.</i>
<i>Outage due to Increase in Main Output Transformer Red Phase Dissolved Hydrogen (Bruce B Unit 6 15, June 2011)</i>	<i>CNA II: Lessons learned were disseminated to Operations staff.</i>
<i>Loss of power to feedwater recirculation valve resulting in a manual reactor trip (Palisades, PWR, USA, December 2011)</i>	<i>CNE: Lessons learned were disseminated to I&C maintenance shop staff</i>
<i>Employee receives electric shock from alternate 600-Volt power supply (Point Lepreau, PHWR, Canada, February 2012)</i>	<i>CNE: Lessons learned were disseminated to Electrical maintenance shop staff. CNA II: Lessons learned were disseminated to Maintenance, Electrical Commissioning and Nuclear Safety staff</i>
<i>Unplanned Shutdown: Failure of Instrument Air Line to Boiler Level Control Valve (Cernavoda Unit 2, PHWR, June 2012)</i>	<i>CNA II: Lessons learned were disseminated to Maintenance staff.</i>
<i>Large component replacements affecting chemistry (INPO IER L4-12- 52, September 2012)</i>	<i>CNA II: Lessons learned were disseminated to chemical systems staff.</i>
<i>Hand injury while closing an airlock door (Darlington 3, PHWR, September 2012)</i>	<i>CNA II: Lessons learned were disseminated to Industrial Safety and Operations staff.</i>
<i>A Leak Observed at the Fire Pump Discharge During Testing of Fire Protection System (Kozloduy 5, WER-1000, February 2013)</i>	<i>CNA II: Lessons learned were disseminated to Maintenance and Industrial Safety staff.</i>
<i>Guideline protection equipment (common event, March 2013)</i>	<i>CNA II: Lessons learned were disseminated to Operations and Maintenance staff</i>
<i>Nuclear Power Plant Software Quality Assurance (Good Practice - common event, March 2013)</i>	<i>CNA II: Lessons learned were disseminated to Operations and Engineering staff.</i>

VI.2. MOST SIGNIFICANT OPERATIONAL EVENTS IN CNA I AND CNE. LESSON LEARNED AND CORRECTIVE ACTIONS RESULTING FROM NATIONAL EVENTS. OPERATING EXPERIENCE

VI.2.1. Manual shutdown due to excessive air leakage from boron injection circuit TB11 (CNA I, September 28, 2010)

The plant was manually shutdown in accordance with the corresponding operating instructions due to excessive air leakage through the Boron injection valve TB11S003 belonging one train of the boron injection shutdown system (TB11). The leakage was detected as a result of the system's compressors start up frequency. The train was available when the plant shutdown was decided.

Direct Cause: Depressurization of line TB11 because of failure in valve TB11S003.

Immediate Actions: Defected valve was replaced. Spare parts assemblies at the warehouse were checked and two of them were disposed.

Root Cause: Valve (TB11S003) not properly assembled with an incorrect sealing ring (seger ring's model). Maintenance only replaces the complete assembly containing the seger ring provided by the supplier.

Lessons learned / Corrective actions:

- *Check valves equivalent valves in other trains (TB21S003 / TB31S003) during scheduled outages. The correct seger rings were found mounted in both cases.*
- *Modify the work plan to check valves of the three trains (TB11/21/31S03) including a quality assurance (QA) stop point to verify that it is mounted with the corresponding seger on piece.*
- *The event was included in the work package to check valves TB11/21/31S03.*
- *The event causes as well as the action taken were discussed in "ad hoc" meetings carried out between the Departments of quality assurance and mechanical maintenance.*

VI.2.2. Automatic SCRAM due to low level on steam generator 2 (CNA I, March 08, 2011)

Due to the failure of pressure gauge (RL11P002) of the SG's feed water tank, two control valves (RA03S002 and RA04S002) opened, releasing live steam flow to the feed water tank. As a consequence, live steam pressure decreased, electric power decreased, pressure and temperature in the feed water tank increased and steam relief valves opened. When reaching an equilibrium status, 10 minutes after, the operator manually closed these valves (RA03/04S002) and, steam pressure increased, steam flow decreased, feed water flow decreased (mainly feed water pump RL51D001) and the SG's water level decreased. Two minutes later a SCRAM occurred due to SG 2 low level.

Direct Cause: Pressure gauge RL11P002 showed incorrect reading. Its power supply source was found burnt out.

Immediate Actions: The plant was taken to hot shutdown condition. The power supply source of the RL11P002 pressure gauge was repaired.

Root Cause: the RL11P002 pressure gauge failed due to failure of its power supply, a capacitor with aging signals was found. The capacitors failure due to aging is an old known problem. Failed equipment is not included in the preventive maintenance programme.

Contributors: A formal programme for replacement of capacitors to prevent its failure due to aging has not been developed. I&C usually checks and replaces the capacitors on its own initiative, even if the task is not in the corresponding Work Order (OT).

Lessons learned / Corrective actions:

- *To prepare a list of the type of modules involved in the event that do not have preventive maintenance, set priorities based on safety and availability and, include such modules in the preventive maintenance programme.*
- *Assess a plant modification taking into account both, redundant pressure measurement in the feed water tank and triggering relief valves.*
- *A new prototype of card was developed which is in its final validation process and trial.*

VI.2.3. SCRAM during manual shutdown (dynamics of shutdown) (CNA I, June 18, 2011)

Because the SG's control valve (RL11S005) had been fully opened and locked after the gasket adjustment carried out as a consequence of water and steam leakage detected and, given the impossibility of repairing it with the plant in operation, a decision was taken to manually lead the plant out of service smoothly and in a controlled way. During this operation, an unexpected close of an isolation valve (belonging to control valve RL11S030) occurred. Under these conditions, the SG 1 level started decreasing and the operator manually carried out a reactor scram, simultaneously with the automatic SCRAM which occurred due to low level signal in SG 1. The SG's level reached the trigger level of the emergency feedwater system (SHS, Second Heat Sink system).

Direct Cause: Unexpected closing of valve RL11S030.

Immediate Actions: The plant was stabilized in hot shutdown condition.

Root Cause: The SHS actuation is expected in this type of transient. High humidity in the environment where is located the valve RL11S030. Moisture in the connection linking reverse signals of the RL11S030 valve's command limit switches with an intermediate box due to failure of its rubber seal.

Lessons learned / Corrective actions: Inclusion of the mentioned transient in the simulator used for operators' training.

VI.2.4. Deposits of deuterated boric acid in crystal form inside the piping of Shutdown System 2 (System deuterated boric acid injection, TB) (CNA I, March 30, 2012) IRS N° 8241

Owing to the loss of deuterated boric acid solution through the weld in venting pipe TB31 Z02 (RT1), during the technical review carried out in 2012, was decided to repair of venting pipe connected to boron injection pipe TB31 Z02 as well as the replacement of the membrane TB31 S011. For this purpose, and as a result of the radiological assessment of the task, it was decided to remove the pipe from the reactor room.

Upon removing the pipe, deposits of deuterated boric acid in crystals form were observed inside it, near the end where the rupture membrane is secured. On account of the discoveries made in branch 3, a decision was taken to investigate the other two branches (TB11 and TB21 Z02) where deuterated boric acid crystals were also found. The crystallization and precipitation of deuterated boric acid is presumed to have taken place at low temperature, a phenomenon that would occur during cold shutdown of the plant. The phenomenon would therefore be accumulative and had probably been going on throughout the CNA I.

Direct Cause: Precipitation of deuterated boric acid.

Immediate Actions:

- Verification of the length of pipe covered in crystals and quantity (weight measurement) of deuterated boric acid. In branch 1, the estimated length was 80 cm; in branches 2 and 3, the estimated length was 40 cm. The weight of the deuterated boric acid crystals was: branch 1: 3650 g; branch 2: 2590 g; branch 3: 1820 g.*
- Permeability testing of the deuterated boric acid (precipitate) in the pipe, through pressurized air injection. It was proven that air could pass through the deposit.*
- Ultrasonic examination of pipes. Mechanical cleaning and, subsequently, chemical cleaning. Visual inspection to confirm that the cleaning was complete.*
- Boroscopic inspection of the deuterated boric acid tanks.*
- Replacement of rupture discs.*
- Installation of the pipes in the system. Ultrasonic non destructive tests (NDT) were performed to determine whether there were variations in their thickness.*

Root Cause:

- Design: The documentation provided by the designer did not include a preventive maintenance or monitoring programme. Some components were not designed for all the environmental conditions of normal operation.*
- Operating status: Effects of operating status corresponding to plant cold shutdown not assessed.*
- Safety assessment: Not fully implemented. The failure modes of one segment of the system were not taken into account.*
- Management: There are no procedures to protect the pipes during cold shutdown.*
- Environmental conditions: Requirements for the interface between some systems (TB/TK) were not taken into account.*
- Maintenance/tests: A monitoring programme has been not implemented throughout the plant's operation. The QA system did not detect the weaknesses revealed previously (maintenance and monitoring) during the plant's operation.*

Lessons learned / Corrective actions:

- Keep the room where the deuterated boric acid solution is stored at a temperature no lower than 20 °C.
- Verification during the plant startup process. The temperature in the pipes will be controlled using thermocouples; when it reaches 20 °C, the deuterated boric acid tanks will begin to fill.
- During the next outage that involves putting the plant in cold shutdown, the reverse will occur. When the pressure in the primary circuit reaches 35 ATA and the temperature reaches 100 °C, the tanks will begin to empty and the concentration of the deuterated boric acid solution inside will be measured to check that it remains at 4%.
- A boroscopic inspection of the pipes will be performed.
- Possibilities are being analysed. A system will most probably be designed that keeps the pipes at a temperature above 20 °C during cold shutdown of the plant. Together with a contractor, a new prototype of card was developed which is in its final validation process and trial.

VI.2.5. Outages due to turbine trip caused by failures in main generator exciter (CNE, May 26, 2010) IRS N° 8169

On 2010-05-26 and 2010-05-29, there were reactor trips as a result of low core flow and high neutronic power. During these incidents, the transfer of the power supply from 500 kV to 132 kV to the bus bars of class IV and class III did not occur automatically (fast transfer) because the voltage fluctuations caused by exciter defects actuated the generator's electrical protection.

The plant was reconnected to the grid on both cases on 2010-05-28 and 2010-06-01, respectively. Since the fault only occurred when the exciter was in operation, it was decided to synchronize on 2010-06-01 with operating power limited to 70% full power (FP) and auxiliary services supplied from the 132 kV system in order to prevent the occurrence of another reactor trip due to main primary pumps shutdown, as occurred on the two previous occasions. On both occasions the various systems functioned as designed and the plant could be returned to service after waiting out the reactor poisoning period.

On 2010-06-02 at 17:22, this fault occurred again, resulting in a plant outage without reactor scram. An automatic setback to 60% FP occurred, as per design. On 2010-06-04, the plant was connected to the national grid under the same operating conditions as in the previous event. On 2010-06-06, with the reactor at 70% FP, another outage occurred which left the reactor at 60% FP.

Direct cause: fault in the main generator exciter which caused fluctuations in the exciter output voltage with consequent alterations in main generator output voltage. These voltage fluctuations activated the minimum voltage protection systems for equipment connected to the 6.6 kV busbars that are essential for plant operation.

Root cause: ageing of the electronic cards components regulating exciter output voltage. At least one of them presented a non-evident fault.

Lessons learned / Corrective actions:

- To remove the recorders used to monitor the behaviour of the exciter (ID 110984).
 - o The modification suggested by Ansaldo to change the exciter logic (in case, voltage regulation being in AUTO, there is a reduction in reactive power and the under-excitation limit is activated) should be made standard, so that, in the event of an under-excitation signal, control is switched over to MANUAL (ID 110984).
 - o In the first two events analysed (reactor at 100% FP), the actuation of shutdown system No. 1 were due to low flow in the main heat transport system and high neutron power. Both these parameters were activated owing to the tripping of a number of loads, including the main heat transport system pumps, as a result of the actuation of the protection system for low voltage in the busbars of class IV and class III. This signal was activated ahead of the low voltage signal which trips the circuit-breaker for the power supply to those busbars from the 22/500 kV system.

With the involvement of experts from Ansaldo and Reivax, the exciter control components were replaced and the fault was eliminated. On 2010-06-07, the plant was reconnected to the national grid with the reactor at 60% FP, and the services continued being supplied from the 132 kV system until

2010-06-12, without incident. The power was increased to 95% FP. On 2010-06-14, the normal power supply was restored for auxiliary services from the 500 kV system and power was then increased to 100% FP.

VI.2.6. Leak of D₂O from a Feeder Channel (CNE, December 29, 2010) IRS N° 8227

Since 8 December 2010, CNE was in planned shutdown to change two pressure tubes of channels Q05 and Q07. To optimize dose, a decision was taken to change both channels at the same time. On 29 December, with the work on changing the channels finished, all that remained to do was the pressure testing and subsequent coupling with the primary heat transport system (PHTS). The Q07 feeders were frozen and drained as far as the T-101 flange in order to begin testing that pressure tube first. The Q05 feeders were not frozen. The work done on Q07 was handed over to the night shift for them to remove T-101 and replace it with flange T1501.

While the tasks were being carried out, a worker erroneously loosened the bolts of the flange T-101 of channel Q05 instead of Q07, thereby causing a significant leak of D₂O (approximately 5,290 L).

Direct Cause: Human error when removing the flange from channel Q05.

Root Causes:

Organization of work: The work methodology used in the mock-up training was not put into practice during the incident, particularly in the following areas:

- *Communication:* the telephone and video camera were not used for each step of the procedure.
- *Supervision:* there was no step-by-step monitoring of the person performing the task.

Lack of safety culture: this was observed at the following stages of the incident:

- *Supervisor was not present and the person providing support had gone to fetch tools at the time when the incident occurred.*
- *The shift radiation protection officer was not informed that work would be carried out simultaneously on both sides of channel Q07.*

Consequences: This incident caused the leak of approximately 5,300 L D₂O, of which was estimated that 250 L were released to the outside through the ventilation system. The maximum dose received by the workers involved in post-incident recovery activities did not exceed 4.5 mSv.

Lessons learned / Corrective actions:

- *Review of work plan procedure applied (PT 1075);*
- *Improve tasks in mock-up training;*
- *Improve practices involving human performance;*
- *Improve organization safety culture.*

VI.2.7. Unable to discharge spent fuel from the refuelling machine during approximately 10 hours (CNE, October 01, 2011)

During discharge of spent fuel assemblies in the side A port of spent fuel, after discharging the first pair, the fuelling machine (M/R) RAM "C" was clogged, which prevented immediate discharge of last three spent fuel assembly pairs.

A number of forward and retraction tests were conducted in the M/R RAM "C" by using forces higher than normal. Finally, the indicator on the control board stated that M/R RAM "C" was moved forward about 2 cm from its expected position and, the pressure and flow were exceeded from the normal values.

Various tests with valves associated to the M/R RAM "C" operation (with D₂O) were conducted, but problems were not solved. Therefore, the rest of the fuel assemblies were bled using M/R RAM "B", for which a special operation procedure was created and, the M/R operator was instructed about it and the spent fuel assemblies were evacuated into the pool.

A mechanical control on M / R RAM C that consisted in checking that RAM "C" stopped after moving forward 15 cm before reaching home position was carried out. Following RAM inspection, RAM assembly was changed. This assembly had only completed 30 cycles, since it had been replaced by preventive maintenance.

Direct Cause: clogging of the M/R RAM C side A.

Apparent cause: Two-lock-screw assembly securing RAM C piston threading was loose. This resulted in friction with the higher section of one of them in the RAM Latch tube inner face, in the last tube portion. This defect is generated because, when these two screws (worms) are drove in, they did not penetrate into the housing on their piston outer face. This, in turn, made the piston jam increasing the assembly friction.

Consequences:

- *Unable to discharge three pairs of spent fuels by normal operation.*
- *RAM assembly had to be replaced before reaching the design operation cycles.*
- *Spent fuel replacement was not performed for 4 days.*

Lessons learned / Corrective actions:

- *The mounting procedure of the RAM assembly was checked, highlighting the piston's safety worms drive-in details and stating an outage period for the supervisor to control this drive-in.*
- *A report was sent to the seals supplier, to check their building material, due to that were found oxide spots.*

VI.2.8. During safe shutdown, liquid areas changed to automatic control as a result of positive power error - PP2011 (CNE, October 25, 2011)

During routine checks, at night operating shift in main control room (MCR), the operator detected that startup flag#1 was not enabled, measuring power at -6.23 decades in ion chambers and -6.136 decades in the logarithmic power indicator (PotLog).

MCR's team decided to activate the startup flag #1. Immediately, disturbance was generated in the liquid area control system. Areas changed directly from "manual" to "auto," increasing their average level.

- *Abrupt disturbances were generated in the helium pressure of the balance collector of the liquid area control system.*
- *Absorber rod banks were partially introduced.*

Next, Operator assumed that a positive power error had been produced and proceeded to push the "Hold Power" button.

After that, the operator checked critical parameters on "Reactor Regulation Condition" screen as well as alarms, finding that reference value for Reactor Regulation System (RRS) was set on 0.0001%. On the other hand, the logarithmic power was stable at -5.904 decades. The operator set a new power reference value at 0.1% to produce a negative error.

The RRS started to decrease the average liquid area level and to withdraw the absorber rods banks. Once absorber rods banks were removed and area level average was at 50%, the operator set liquid area level reference value in "manual" mode at 50%.

Direct Cause: positive reactor power error, which generated liquid area level control transference from manual to automatic mode.

Apparent Cause: error when enabling startup flags in DCCX control computer due to lack of both the suitable training and a procedure to perform this handling. The corresponding instructions are poorly detailed (MO-63700 Operating Manual Instructions, section 4.3.9.).

Consequences: Temporary disturbance in the reactor power control mode during safe shutdown without consequences.

Lessons learned / Corrective actions:

Introduce an addendum to the Operating Manual Instructions which includes the procedure to enable the startup flags in safe shutdown and below -6 decades according to the scheme indicated below, considering that the controlling computer is DCCX.

VI.2.9. Explosive mixture ignition in steam generator during opening process for inspection - PP2011 (CNE, November 13, 2011)

On 13/11/2011, at approximately 6 p.m., following the removal of the inner cover (entrance of man to calotte) of the SG#4, heavy water buildup at the bottom to the cold leg began to be drained.

For do that, a self-priming centrifugal pump was being used with variable speed device, powered by 220 V AC, and operated by a worker from outside through manhole. In this operation, pump body was placed outside the SG, when the suction began and the water level decreased, a suction pipe was introduced to the calotte center. So pump motor was getting closer and closer to the manhole opening. After operating the pump for approximately 15 minutes, a deflagration occurred inside it, with a bluish flame which was spread through the manhole to the outside, affecting the worker that performed the task.

Direct Cause: Pump motor collector with which the calotte of the SG#4 was being drained, caused the ignition of a hydrogen-air explosive mixture in the inside, with the described consequences.

Apparent Cause: Lack of awareness of hydrogen presence in primary heat transport system (PHTS) when the SGs are drained to the half of the PHTS' headers level.

Consequences:

- Burn injuries in worker's torso, hands, arms and right leg.*
- Contamination (low level) in worker's abdominal area.*
- No damages on the installation's SSCs.*

Lessons learned / Corrective actions:

- Preparation of the transfer pump, lights and wiring suitable for explosive atmospheres to introduce them into the SG's calotte.*
- When manhole covers in GVs are removed, explosive mixture in calotte and in working area must be measured and must be installed, a ventilation system to suck out the SG's inner by removing the existing air/hydrogen mixture through the D₂O steam recovery system.*
- Modify the Work Plan 1054 Annex III Radiation Protection Instructions including the above mentioned corrective actions.*

VI.2.10. Wrong 3211-PM2 motor exchanger replacement (CNE, April 02, 2012)

When beginning the replacement task of one of the two heat exchangers (HX) of the moderator main pump motor (3211 PM2) due to a HX's tube leakage, which was identified by means of a drawing copy (flowsheet) of the motor cross section. The HX's drilled tube was identified as related with the level switch (LS) 63211 LS64#1. This LS and the 3211 PM2 motor are at inaccessible locations under normal plant operation. When the replacement was over, it was confirmed that the leak continued.

Therefore, after analyzing again the issue, it was determined that, although in the drawing is stated that tube leaks were collected in 63211 LS 64#1, in the field, the LS that collected the disassembled exchanger leaks was identified as 63211 LS 64#2. So, incorrectly the exchanger without leaks was replaced.

Direct Cause: Crew involved in the event have wrongly identified the exchanger to be replaced.

Apparent Cause: Poor use of human performance tools by the involved personnel belonging to operation, mechanical maintenance and Quality Assurance. Lack of critical attitude, self-check and independent checking.

Consequences:

- *Loss of time and need of a new planning, scheduling and performance that had to be planned as a work of the planned outage. The task had to be performed through complicated alternative methods due to lack of a bridge crane that was under maintenance.*
- *Unnecessary dose in the crew involved in the task. These motors were located in the moderator room in which there was a hot spot very close to the replaced exchanger. To lower the dose for the crew, lead plates had been placed. However, contact exposition at that point was 0.1 mSv.*

Lessons learned / Corrective actions:

- *Report the event to the crew involved: mechanical maintenance , operations, quality assurance and, working package preparation.*
- *Modify and verify the operating flowsheet 320001-2 OF for 3211 PM 1 and PM 2 pumps.*
- *Reinforcement of the performance tests among the crew involved in the task: mechanical maintenance , operations, quality assurance and work preparation.*

VI.2.11. 7131-PM3 seal damaged due to lack of water when applying a working protection (CNE, July 12, 2012)

On 12/07/2012 a working protection (WP) was applied to a pump (7131-PM2) of the low pressure process (service) water system to prepare the motor's heat exchangers. When applying the WP, the #3 process pump alarms for low cooling water level and low seal water level were triggered on main control room. Due to this situation, the field crew were required to undo the last steps taken and the alarms were normalized.

It was checked that the 71115 V25 valve cut the water supply to the 7131-PM3 motor cooler (active), instead of doing it to the 7131-PM2 motor cooler.

Besides, the field crew detected smoke on the seal of the pump 7131-PM3, reported this situation and, it was decided to suspend the WP related to 7131-PM2 pump, then this pump was put in service and, the 7131-PM3 pump was taken out of service.

Direct Cause: cooling water interruption to the pump seal.

Apparent Cause: incorrect identification of the 71115 V25 valves.

Consequences: The seal of the 7131-PM3 pump was damaged when the water supply to the motor coolers was closed, impeding the water supply to the pump seal.

Lessons learned / Corrective actions:

- *Identify correctly, in the field, the valves supplying water from the seal to the pumps 7131 PM2 and 7131 PM 3.*
- *Create an work instruction to check the disassemble of SSCs and include it in the pack of the corresponding working guidelines.*

VI.2.12. Service outage due to the simultaneous closure of valves 41113 MV1, MV2, MV3 and MV4 during functional test PF 41100-07 (CNE, May 06, 2012)

On the 2011 planned outage, the analogical logic of the turbine's electrohydraulic control system (EHC) has been replaced by a new digital logic named digital electrohydraulic control system (DEHC).

On May 6th, 2012 at 4.46 p.m. with the plant operating at 83% FP, the functional test (PF-41100-07) of the turbine's main emergency valve 41113-MV2 (SV2) was performed. During the test, an electronic board failed when opening the valve. Consequently, this valve remained 20% open during more time than the allowed period for the DEHC logic and has been enabled the turbine's heating interlock which demanded the closure of the valves 41113-MV1, MV3, and MV4. Due to this situation, the steam pressure has increased to reach the initiation set-back value for this event. Afterwards, the turbine was manually triggered and the reactor power was increased up to 58% (5:13 p.m.). The damaged electronic board was replaced and the turbine set was synchronized with the electric power network at 8.45 p.m.

Owing to the heating control logic was disabled when began the test and, considering that a timer has been used to set the return of the valve to the wide open position to complet the test, it was determined that the same logic of the EHC system original of the turbine sets the end of the test in the same stroke end, initiating the interlock for closing the valves 41113-MV1, MV3 and MV4. Therefore, it is recommended come back to the test end instruction by the mentioned stroke end.

Direct Cause: partial closure of the valve 41113-MV2 and total closure of the valves 41113-MV1, 41113-MV3, and 41113-MV4.

Apparent Cause: change of the high pressure heating control logic of the turbine and the DEHC that replaced the original EHC.

Consequences: Disconnection of the network generator and electric power loss.

Lessons learned / Corrective actions:

Include in the work instruction (ID:115279) regulation the change of the high pressure turbine heating control logic and the emergency main valves returning to the instruction of test end by using the stroke end that indicated 90% opening of the valve 41113-MV2 eliminating the opening timer.

VI.2.13. CNE alert situation caused by trespassing (CNE, March 11, 2013) IRS N° 8313

On 11 March at 10:26 a.m., a thirty eight persons group belonging to the environmentalist group Greenpeace entered to the Embalse site. They got in through a door that was open due to a painting work that was being carried out in an adjacent building. Their entry was detected by means of closed-circuit television cameras, and the security staff notified the plant manager, the safety and radiation protection officer and the National Gendarmerie (military-type public security force designated as a response force for physical protection of strategic places). The security staff set off the intruder alarm and the relevant actions set out in the emergency plan were initiated.

The National Gendarmerie was called to remove the trespassers and the safety-related buildings were sealed off (including the main control room and the controlled area).The operator suspended the activities planned in the spent fuel dry storage system.

The environmentalist group continued on its way, through a second checkpoint consisting of vehicle control barriers and, ignoring the warnings of the National Gendarmerie and the security staff, headed towards the stairs leading to the roof of the services building and the roof of the reactor building. Some members of the group went up onto the roof of the reactor building, while the rest remained tied by themselves to the building's staircases, blocking them at intermediate levels and at ground level to prevent the security staff's access to the roof. Once they reached the reactor building roof, they displayed signs with anti-nuclear slogans. At the same time, security staff saw that two boats were approaching on the lake. National Gendarmerie staff were notified of this situation and they sent guards to secure the area. The boats were kept away from the shore.

The plant manager asked the commander of the National Gendarmerie to carry out the group's move, which was done with some delay because authorization was required from the federal judge with jurisdiction over the power plant area. Finally, at around 12:20 a.m., the group was removed by the public security force and handed over to the authorities. None of these persons gained entry to the power plant buildings.

The nuclear safety of the facility was not affected, nor was the physical protection of material or persons. The following actions were taken during the event:

- Detection of the intrusion by means of closed-circuit television cameras.
- Activation of the intruder alarm and implementation of the emergency plan.
- Interception of the intruder group by facility security staff.
- Notification of access control and security operations supervisors.
- Notification of the plant manager, safety and radiation protection officer and public security force.
- Suspension of activities planned in the dry storage spent fuel system.
- Access blocking to safety-related buildings.
- Attempt to stop persons which climbed onto the reactor building roof.

- Detection of boats approaching on the lake and protection of the potential access area.
- Removal of the intruder group by the public security force.

Actions in the short term:

- The responsible entity strengthened security at nuclear power plant sites.
- The Regulatory Body informed officials at other facilities guarded by the National Gendarmerie about the event.
- An internal operating experience procedure was launched to identify the causes of the incident and the corresponding corrective actions.

VI.2.14. Other operational events in CNA I and CNE. Lesson learned and corrective actions resulting from national events and operating experience

In the following chart other events that occurred in CNA I and CNE during 2010 – 2013 are given with less details.

EVENT / OPERATING EXPERIENCE	EXAMPLES OF LESSONS LEARNED / CORRECTIVE ACTIONS
<i>Manual SCRAM due to failure at normalization after a repetitive test (CNA I, April 09, 2010)</i>	- A new card prototype was developed which is in its final validation process and trial. - A test plan will be developed for this prototype card.
<i>Manual shutdown due to excessive air leakage through a pilot valve of TB31S003 (CNA I, December 17, 2010)</i>	- Include the replacement of all pilot valves' sealing elements by new elements after each intervention in internal Instruction IT 20, after each checking. - Determine a methodology to identify and keep the history of each pilot valve, regardless of the valve where they are mounted. - All sealing materials of the pilot valves were removed and disposed from the I&C workshop. - The drive piping of the Boron injection valves were cleaned.
<i>Manual Shutdown due to failure on the alternator seal oil system (CNA I, May 20, 2011)</i>	- Include in the preventive Maintenance of the separators tanks SU11B501 and SU11B502. - The floating design was changed for another one, in which the operation is not affected by oil entering.
<i>Load rejection due to manual opening of line switch from the local control (CNA I, June 27, 2011)</i>	- The Operating Division discussed this issue by stressing the use of human performance techniques and the operation of the local board of switch AC04. - Internal instruction to reset the AC04 switch disturbance due to low oil pressure. - Install a telephone at the local board of switch AC04.
<i>Manual shutdown due to main generator differential protection actuation (CNA I, February 07, 2012)</i>	- Replacement of current transformer TI2 AP01f11, working with the designer support during the scheduled outage 2012. - All toroidal current transformers of the main generator's protection system (SP) were revised, with the designer support during the scheduled outage 2012.

EVENT / OPERATING EXPERIENCE	EXAMPLES OF LESSONS LEARNED / CORRECTIVE ACTIONS
<i>Failure on Block Transformer AT01 (CNA I, March 27, 2012)</i>	<i>The failed transformer was replaced. A multi gas analyzer on Siemens transformer was installed.</i>
<i>DCCX computer service outage during air conditioning installation (CNE, September 06, 2011)</i>	- <i>Event occurrence report among maintenance group personnel involved belonging to electrical maintenance and to computer maintenance.</i> - <i>Practice on a daily basis for the use of equipments, supports, cabinets, etc; as an elevation tool to perform the tasks.</i> - <i>Avoid the use of any kind of electrical tools (drills, vacuum cleaners, etc.) to work in the computer room, with the computers in operation.</i> - <i>Maximize actions so that only the use of pneumatically driven machines is allowed to work in the computers room, performing these tasks so far as possible with computers out of service aiming to minimize vibrations and/or noise.</i>
<i>Contamination spread during cobalt-60 bars charge / handling (CNE, October 20, 2011)</i>	- <i>Follow the Procedure</i> - <i>Modify the procedure PI 1048 by including the following indications:</i> • <i>Install a portable suction sleeve to ventilation system as to force the venting of the shielding when sweeping with gas;</i> • <i>During sweeping operation, the involved personnel must use breathing protection and rubber gloves.</i> - <i>Report the event</i>
<i>Opening failure of the valve 3432-MV71 belonging to high pressure emergency core cooling system (CNE, December 13, 2011)</i>	<i>Check of valve actuators related to 3432 MV 71, including valves 3432 MV 72, 3432 MV 79, and 3432 MV 80.</i> <i>Report this event in Electrical Operation and Maintenance areas.</i>
<i>D₂O spillover due to loss in supply actuators of refueling machine (M/R) "A" (CNE, July 10, 2012)</i>	- <i>Train the crew involved in controlling the seal condition every time a new hose coupling should be attached.</i> - <i>Include in preventive maintenance system the control of all the couplings every time an inspection sheet (PI) is requested to take a hose off line.</i>

ANNEX VII

RESUME OF NA-SA QUALITY ASSURANCE

GENERAL MANUAL CONTENT

VII.1. TABLE OF CONTENTS

Introduction (objective, scope, grading requirements, distribution).

Quality Policy.

Basic Requirements:

- Documentation.
- Direction: organization, responsibilities, training, corrective and preventive actions, controls.
- Execution.
- Evaluation: self assessment – independent evaluation.

Annex 1 Nucleoelectric Argentina S.A. General Organization Chart.

Annex 2 Procedures & General Documents.

Annex 3 Definitions / glossary.

VII.2. QUALITY POLICY

Nucleoeléctrica Argentina S.A. (NA-SA), an organization committed to electric generation in a safe, competitive and clean way, declares and assumes the following Quality Policy:

- **Ensure the control of activities**
Perform a continuous effort to plan and control every activity directly or indirectly related to safety and the availability of its facilities, applying the appropriate codes and technical standards throughout the Company.
- **Adapt the management to the applicable standards**
Comply with applicable standards to facilities and activities.
- **Promote personnel training**
Train personnel ensuring its competence in order to develop the assigned tasks, conscious of the safety issues.
- **Promote internal and external communications**
Communicate the Quality Policy to all the personnel, ensuring its understanding and fulfilment.
- **Continuously improve Quality Management**
Try to achieve continuous improvement by the systematic and periodical assessment of quality management, and by benchmarking with the best international practices.

VII.3. ENVIRONMENTAL POLICY

Nucleoeléctrica Argentina S.A., an organization *dedicated* to electric generation in CNA I, electric generation and Cobalt production in CNE, maintenance construction and commissioning in CNA II *Project, activities related to Embalse NPP Life Extension Project; administrative activities related to a IV NPP Project, Technical and Administrative Management at Headquarters*, declares and assumes the following Environmental Policy:

- **Prevent Environmental Contamination**

Perform a continuous effort to prevent and diminish the adverse environmental impact resulting from our activities in nuclear power generation and operate our plants making a rational use of energy and natural resources.

- **Continuously Adapt Environmental Management to the Applicable Standards**

Comply with the applicable standards to facilities and activities and any other requisite subscribed by the organization.

- **Promote Personnel Training in Environmental Care**

Train personnel ensuring its competence in order to develop the assigned tasks, conscious of the safety and environmental issues.

- **Promote Internal and External Communications**

Communicate the Environmental Policy to all the personnel, ensuring its understanding and fulfilment and make it available to all parties involved. *Inform* the clients and general public on the benefits of the nuclear option and its contribution to environmental preservation.

- **Evaluate Potential Risks of the New Projects and minimize Environmental Impact during its execution**

- **Continuously improve *Environmental* Management**

Achieve continuous improvement by the systematic and periodical assessment of environmental management.

ANNEX VIII

REGULATORY REQUIREMENT RQ-NASA-38

COMPREHENSIVE SAFETY EVALUATION OF THE

ARGENTINEAN NUCLEAR POWER PLANTS

(STRESS TESTS)

VIII.1. REGULATORY REQUIREMENT RQ-NASA-38

The Nuclear Regulatory Body (ARN) requires to Nucleoeléctrica Argentina S.A. (NA-SA) fulfil the following regulatory requirement:

Send to ARN, before 04/27/12 for CNA I / CNE and before 05/31/12 for CNA II, a report containing a Comprehensive Safety Evaluation of these NPPs that, complying with what is specified in the attached Annex, includes:

1. The extreme initiating events conceivable at each NPP site.
2. The loss of safety functions caused by each one of the extreme initiating events considered.
3. The severe accident management corresponding to each one of the extreme initiating events considered.

The Comprehensive Safety Evaluations requested should include:

- a) The long-term evolution of severe accidents (referred to in point 3) caused by extreme events (referred to in point 1) and the ability to recover / repair the supplies of cooling water and electricity until a stable NPP condition is achieved, in order to identify the more suitable strategies for repair and recovery as well as the elements that have to be available for their implementation. Indicate the expected schedule to have the above mentioned elements available.
- b) The safety implications resulting from the existence of multiple reactors at the Atucha site in case of extreme events (listed in point 1), identifying and implementing the appropriate measures that eventually must be adopted and the procedures to use the existing resources from one unit for the other one. Indicate the corresponding schedule when this would be achievable.
- c) Spent fuel management strategy (SF) and the design and the performance of the SF storage systems in case of extreme-event occurrences (referred to in point 1). Determination of the foreseen actions and the schedule for their implementation.
- d) Disposal of equipment and components belonging to safety systems and safety-related systems to ensure that, in case of extreme-event occurrences (such as those listed in point 1), those systems can continue performing the function for which they were designed. In the case it were necessary to implement modifications, indicate the planned measures and the corresponding implementation schedule.
- e) Measures for prevention, recovery and mitigation of accidents characterized as indicated in items 2 and 3. Reassess both the automatic and the operators' actions established in the operating procedures for abnormal conditions, emergency and severe accident management. Indicate the foreseen schedule for the implementation of the improvements eventually decided on.
- f) Availability of the existing resources in the nuclear power plants in response to internal and external nuclear emergencies in case of accident scenarios characterized as described in points 1, 2 and 3. Furthermore, during the time that elapses between the accident occurrence until the ARN assumes the responsibility for the emergency management, the planning and the management actions must be foreseen including the communications and the protection of the public. For each of the above issues, the necessary improvements should be identified, and their implementation schedule given.

VIII.1.2. Annex to Regulatory Requirement RQ-NASA-38: Specifications for the Comprehensive Safety Assessment that shall be Carried out by the Nuclear Power Plants

VIII.1.2.1. Comprehensive Safety Assessment Definition

The Comprehensive Safety Assessment (stress-tests) is aimed to determine the nuclear power plants safety margins considering extreme events such as those that occurred in Fukushima, which may endanger the safety functions of the NPPs and may lead to a severe accident situation. This assessment should include:

- An analysis of the response of each NPP in case of the occurrence of a set of extreme situations as considered in item VIII.1.2.4. of this Annex, and;
- The verification of the prevention and the mitigation measures based on the defence-in-depth concept.

In the aforementioned extreme situations, - the sequential loss of the existing defence-in-depth lines must be analyzed -under a deterministic approach-, independently of the likelihood of such a loss. It should be considered that the loss of safety functions and severe accident situations can only occur when numerous design provisions have failed. Furthermore, it should be assumed that the available measures to properly manage these situations were not successful.

The Stress-Tests shall include for each NPP, the plant response and the effectiveness of the preventive measures, highlighting any potential weaknesses and any cliff-edge-effect identified in the analysis. These cliff-edge effects are defined by limit values of critical parameters where by a small change leads to a disproportionate increase in the accident consequences (these conditions may correspond, for example, when a significant flooding begins when a level exceeding the design considerations is surpassed, or the battery capacity is depleted during a total loss of alternating current - AC power event).

The purpose of the aforementioned is to assess the robustness of the defence-in-depth philosophy, the suitability of the accident management measures, and to identify the potential to implement safety improvements, both technical and organizational, such as procedures, human resources, emergency response organization or use of external resources.

The Stress-Tests should focus on such measures that may be taken after the loss of safety functions in case of accidents already considered in the design. Furthermore, the hypothesis relating to the operation of such systems must be re-evaluated. Finally, all measures taken to protect the integrity of the fuel elements in the reactor or in the spent fuel storage systems (wet and dry) should be considered as an essential part of the defence-in-depth.

VIII.1.2.2. Scope of the Stress Tests

The technical scope of the stress-tests was defined considering the problems caused by the events that occurred at Fukushima, which included a combination of extreme initiating events and multiple failures. Therefore, the following situations should be considered:

- a) Conceivable initiating events at the plant site during extreme meteorological conditions:
 - Earthquakes.
 - Flooding / low water level (river / lake).
 - Other extreme natural events.
- b) Loss of safety functions:
 - Loss of electrical power.
 - Loss of the ultimate heat sink.
 - Combination of both of the above.
- c) Severe accident management issues. Measures to manage the loss of:
 - Core cooling function.
 - Spent fuel cooling function.
 - Containment integrity.

d) Considerations about the internal emergency management.

- Direction and Control.
- Fuel damage mitigation.
- Radioactive releases reduction.
- Procedures review.
- Equipment.

The following defines the general information and issues to be considered by NA-SA for each extreme situation.

VIII.1.2.3. General Aspects

VIII.1.2.3.1. Stress-Test Report Content:

Considering that the Stress-Test Report shall be submitted to other institutions for their consideration, the report should include the following basic information for each facility:

- Details of the site location
- Number of units
- Reactor features
- Brief description of the systems involved in the required analysis and that of the existing heat sinks, and the corresponding heat removal chains.
- Date of first criticality
- Type and storage location of spent fuel.
- Significant safety differences between the units.
- The scope and main results of the available Probabilistic Safety Assessments -PSA- shall be provided. If applicable, indicate the use of APS in the evaluation.

Each of the situations included in the Scope (item VIII.1.2.2) and in the Content shall be assessed following the indications given below.

VIII.1.2.3.2. Hypothesis

For existing plants, the assessments shall refer to the current state of each power plant, as built and / or operated on July 30, 2011.

The approach should be essentially deterministic, i.e. when analysing an extreme scenario, a progressive approach shall be followed, in which protective measures are assumed to be lost sequentially during a prolonged period.

The initial plant conditions should represent the operational states of the plant at full power and unpressurized cold shutdown.

It is presumed that all reactors and fissile material or waste facilities are affected at the same time. The possibility of degraded conditions surrounding the NPP, characterized by the lack of external electrical power, difficulty in accessing it, etc. should also be considered.

Besides, consideration should be given to:

- Automatic actions;
- Operator actions specified in emergency operating procedures;
- Any other planned measures for the prevention, recovery or accident mitigation.

VIII.1.2.3.3. Information to be included in the report

The following main aspects must be included in the stress-test reports:

- Provisions taken in the design basis of the plant.
- Plant robustness beyond its design basis. For this purpose, the robustness of the structures, systems and components (SSC) relevant for safety and the defence-in-depth preservation must be assessed (available design margins, diversity, redundancy, structural protection, physical separation, etc.). Regarding the installations robustness and the available measures, one of the approaches of the review is to identify the possible cliff-edge-effects which could lead to a

relevant change in the sequence of events and the existing measures to avoid extreme conditions.

- Possibility of implementing modifications to improve the present defence-in-depth level.

The existing protective measures aimed at avoiding the extreme scenarios considered in the assessment scope must to be described. The analysis should be complemented, where necessary, by the results of the dedicated plant walk-downs performed.

In this regard, NASA should identify the measures to maintain the three fundamental safety functions (control of reactivity, fuel cooling, confinement of radioactivity) and support functions (electrical power supply and cooling), taking into account the probable damage done by the initiating event as well as any available measure to face the event even if they are not considered in the plant safety analysis.

In addition, NASA should identify the possibility of having mobile external equipment and the conditions of their use; the availability of alternative sources of coolant supply; any existing procedure to use resources from one reactor to help another reactor and the dependence of safety functions of one reactor on other reactors on the same site.

Regarding severe accident management, NA-SA must determine, when relevant, the time interval before damage to the fuel becomes unavoidable. If the fuel is in the reactor core, the time available before the water begins to leave the fuel exposed should be indicated as well as the time before fuel degradation initiates (the start of fast cladding oxidation with hydrogen production). If the fuel is in the spent fuel pool, the time before pool boiling commences, the time during which an adequate shielding against radiation is maintained, the time before the water level reaches the top of the fuel elements and the time before fuel degradation starts should be indicated (all of these situations without considering operator intervention).

VIII.1.2.4. INITIATING EXTERNAL EVENTS

The stress-test report should include the information detailed in sections VIII.1.2.4.1., 4.2. and 4.3.

VIII.1.2.4.1. Earthquakes

VIII.1.2.4.1.1. Design basis

VIII.1.2.4.1.1.1. Earthquakes against which the plant is designed:

- Level of the design basis earthquake (DBE) expressed in terms of peak ground acceleration (PGA) and reasons for the choice;
- Indicate if the current DBE is different from the DBE taken into account in the original licensing basis ;
- Methodology used to evaluate the current DBE (return period, past events considered and reasons for their choice, margins added, etc.), the validity of data with time.
- Conclusion on the adequacy of the design basis, considering the state of the knowledge to date.

VIII.1.2.4.1.1.2. Provisions to protect the plant against the DBE

- Identification of the key structures, systems and components (SSCs) which are needed for achieving a safe shutdown state, defined as the state of the plant in which the long term core sub-criticality, the core cooling and the confinement of radioactive material are guaranteed.
- Main existing operating provisions (including emergency operating procedures, mobile equipment, etc.) to prevent damage to both the reactor core and spent fuel after the earthquake, assuming the unavailability of such SSCs;
- Indicate the earthquake's potential indirect effects, including:
 - o Failure of SSCs that are not designed to withstand the DBE and that, in losing their integrity, could cause the consequential damage of the SSCs that need to remain available (e.g. fire, explosions or pipe breaks that can affect the performance of safety functions, including the action of the operation staff).
 - o Loss of external electrical power supply.
 - o Situation outside the plant, including preventing or delaying access of personnel and equipment to the site.

VIII.1.2.4.1.1.3. Plant compliance with its licensing basis:

- General process to ensure compliance (e.g., periodic maintenance, inspections or testing).

VIII.1.2.4.1.2. Other considerations

- Activities and specific verification tests already started by the plant after the Fukushima accident.

VIII.1.2.4.1.3. Evaluation of the margins

VIII.1.2.4.1.3.1. Severity of the earthquake

- Based on the available information (seismic margin assessment or other engineering studies), an estimation must be made of the range of earthquake severity above which the loss of fundamental safety functions or severe damage to the fuel (in the core or in storage) is not guaranteed.
- Indicate if any provisions can be envisaged to prevent these cliff-edge-effects or to increase the robustness of the plant (hardware modifications, procedure modifications, organizational provisions, etc.).

VIII.1.2.4.1.3.2. Maximum earthquake that the plant could withstand without losing the containment integrity.

- Based on the available information (seismic margin assessment or other seismic engineering studies) analyse what is the severity range of the worst earthquake that the plant could withstand without losing confinement integrity (containment and fuel pool integrity).

VIII.1.2.4.1.3.3. Earthquakes exceeding the DBE causing flooding or downspouts that exceed the design basis flooding or downspout level.

- Indicate if this scenario is physically possible considering the NPP location. To this end, identify if potential serious damages to the structures located inside and outside the site (such as dams or buildings and structures of the plant) could have an impact on the safety of the installation.
- Indicate the weaknesses and failure modes that could lead the NPP to unsafe conditions, identifying any potential cliff-edge-effect and the buildings and equipment that would be affected.
- Indicate whether additional measures can be anticipated to avoid the cliff cliff-edge-effects identified or to increase the plant's robustness (equipment modifications, changes in procedures, organizational arrangements, etc.).

VIII.1.2.4.2. Flooding / low water level (river/ lake)

VIII.1.2.4.2.1. Design Basis

VIII.1.2.4.2.1.1. Flooding / low water level against which the plant is designed:

- Level of the flood / low water level design basis and reasons for their choice.
- Indicate if the level of the current flood / low water level design basis is different from that postulated in the original licensing basis.
- Methodology used to assess the current flood / low water level design basis (return period, past events considered and the reasons for his choice, possible margins added, etc.) and the validity of the data with time.
- Conclusions on the adequacy of the current design basis, considering the state of knowledge to date.

VIII.1.2.4.2.1.2. Provision to protect the NPP from the flood / low water level design basis

- Identification of the critical SSCs which are necessary in order to achieve a safe shutdown condition and are supposed to remain available after the flooding / low water level, including provisions to maintain:
 - o The water intake function.
 - o Emergency electrical power supply.
- Identification of the main design requirements to protect the site against flooding / low water level (platform levels, etc.), indicating the associated monitoring programs.
- Main planned measures foreseen to mitigate the effects of floods / low water level, including emergency operating procedures, mobile equipment, flood / downspouts monitoring, warning systems, etc., to warn of flood / downspout and facilitate their effects mitigation, indicating the associated monitoring programs.
- Verify that the possible effects associated with the flood / low water level have been taken into account, including:
 - o Loss of external power supply.
 - o Situation outside the plant, including those problems that might prevent or delay the access of the personnel and the required equipment to the site.

VIII.1.2.4.2.1.3. Plant compliance with the current licensing basis

- Process to ensure its compliance, for example, periodic maintenance, inspections or tests.

VIII.1.2.4.2.2. Other considerations

- Verification activities and specific tests already initiated by the plant after the Fukushima accident.

VIII.1.2.4.2.3. Evaluation of the margins

- Based on the available information (including engineering studies), indicate the level of the flood / low water level for which the loss of the fundamental safety functions, or serious damage to the fuel (in the core or the spent fuel storage facility) is not guaranteed.
- Depending on the available time between the warning and the flood / low water level, indicate whether additional protective measures can be foreseen and implemented.
- Indicate which are the weak points and identify any potential cliff-edge-effects, identifying the buildings and equipment that the flood / low water level might affect first.
- Indicate if additional measures can be foreseen to prevent the identified cliff-edge-effects or to increase the robustness of the plant (hardware modifications, procedures modification, organizational provisions, etc.).

VIII.1.2.4.3. Other natural extreme events

Extreme weather conditions (storms, torrential rainfalls, tornadoes, etc.). It should be indicated:

- Events and combinations of events considered and the reasons for their selection.
- Which are the weak points and the cliff-edge-effects. Identify buildings and equipment that could be affected.
- Provisions to avoid the cliff-edge-effects above mentioned or to increase the NPP's robustness (design changes, modification of procedures, organizational provisions, etc.).

VIII.1.2.5. LOSS OF SAFETY FUNCTIONS

The impact on plant safety functions resulting from the loss of electrical power and the loss of the ultimate heat sink (UHS) should be analyzed.

VIII.1.2.5.1. Loss of electrical power

The electrical alternating current (AC power) sources to be considered should be:

- Off-site power sources (electrical grid).
- Plant generator.
- Backup generators (diesel generators).
- Other alternative backup sources (diesel generators, water turbines, gas turbines, etc.).

The analysis must assume the occurrence of the sequential loss of these electrical supplies.

VIII.1.2.5.1.1. Loss of off-site power (LOOP)

In this case it must be presumed that the off-site power is lost for a long time and that the site will remain isolated for 72 hours from the possibility of receiving heavy supplies by any transport means, although it is assumed that light mobile equipment could arrive to the site within 24 hours from the event occurrence. The following aspects should be considered:

- Describe how the plant design has taken into account the LOOP as well as the internal electrical power supply systems designed to deal with it.
- Indicate how long the above mentioned internal power supplies could run without any external support.
- Indicate what actions are needed and foreseen to extend the internal power supply equipment's operating time (for example, diesel generators fuel tanks refilling, etc.).
- Identify possible measures to be taken to increase the plant's robustness, such as systems modification, procedures changes, organizational arrangements, etc.

In order to clarify this issue, it must be indicated which systems (such as hydraulic turbines, pumps driven by engine / turbine, etc., systems with stored energy in compressed air tanks, etc.) may be considered available.

VIII.1.2.5.1.1.1. Loss of off-site and on-site backup power sources (SBO, Station Black Out).

In these analyses the following two situations must be considered:

- LOOP + loss of the backup power supplies (backup generators);
- LOOP + loss of the backup power supplies + loss of any other alternative backup source.

For each one of these situations, NA-SA must:

- Provide information on the measures considered in the design for these accident scenarios.
- Provide information on the battery capacity and duration. Analyze the consequences of their total loss.
- Indicate how long the plant can withstand an SBO without any external support before severe damage to the fuel becomes unavoidable.
- Indicate the foreseen external actions to prevent fuel damage, considering:
 - o Equipment already present on site, e.g. equipment from another reactor;
 - o Equipment available off-site or near-by power stations (e.g.: gas turbine or water turbine) that can be aligned to provide power via a secure direct connection; assuming that all units on the site have been damaged,
 - o Time necessary to have each of the above systems operating.
 - o The availability of competent human resources to make these connections (considering their exceptional characteristic).
- Identification of potential cliff-edge-effects that could occur, indicating the additional measures that could be incorporated to avoid their effects or to increase the robustness of the plant (system modifications, procedure changes, organizational arrangements, etc.).

VIII.1.2.5.1.1.2. Loss of the ultimate heat sink (UHS)

The UHS is the mean where the residual heat from the reactor is ultimately transferred. Consideration should be given to the sequential loss of these available sinks needed to cool the reactor and the spent fuel pool under any circumstances.

It should be assumed that the functionality of the various existing heat sinks are successively lost and the site will remain isolated for 72 hours with regard to the possibility of receiving heavy supplies by any transport means, although it is assumed that light mobile equipment could arrive at the site 24 hours after the start of the event.

In this regard, a description of the existing design basis provisions to prevent the loss of the various sinks should be provided. For example, diverse water intakes located in different places, etc.

For these scenarios, it should:

- Indicate for how long the plant could withstand the situation without external support before severe fuel damage is unavoidable.
- Provide information on the existing provisions taken in the design basis and the internal actions foreseen for each of the above-mentioned scenarios.
- Indicate the external actions foreseen to prevent fuel damage, considering:
 - o The equipment already existing on-site, for example, equipment from another unit, etc.
 - o Available off-site equipment, assuming that all reactors at the site have also been damaged.
 - o The time required for these systems to be operating.
 - o The availability of competent human resources to perform the necessary actions (considering their exceptional characteristic).
- Identify the possible cliff-edge-effects which could occur, indicating the additional measures that could be incorporated to mitigate their effects or to increase the robustness of the plant (hardware modifications, procedure changes, organizational provisions, etc.).

VIII.1.2.5.1.1.3. Loss of the heat sinks with SBO

The following information should be provided:

- Indicate for how long the plant can withstand a loss of all heat sinks with SBO without any external support and before severe damage to the fuel is unavoidable.
- The existing design provisions and the internal actions foreseen for each of the above-mentioned scenarios.
- Indicate the external actions foreseen to prevent fuel damage considering:
 - o The equipment already present on- site, e.g. equipment from another unit;
 - o The off-site available equipment, assuming that all reactors in the site are equally damaged
 - o The necessary time to have these systems operating at the site;
 - o The availability of competent human resources to carry out the necessary actions (considering their exceptional characteristics);
- Identify the cliff-edge-effects that could occur, and when they could occur, indicating the additional measures that might be incorporated to prevent their effects or to increase the plant's robustness (hardware modifications, procedure modification, organizational provisions, etc.).

VIII.1.2.6. SEVERE ACCIDENT MANAGEMENT:

The mitigation measures should be mainly considered, even if the event occurrence probability is very low. In this evaluation it should be deterministically assumed that a severe accident occurs.

VIII.1.2.6.1 Describe the accident management measures currently available for protecting the reactor core during the various stages of a loss of cooling function scenario:

- Before occurrence of fuel damage in the reactor pressure vessel, indicating:
 - o Whether there are means to prevent fuel damage for primary high-pressure sequences in the primary heat transport system (PHTS), in case the capability to depressurize the PHTS is not available.
 - o The last-resource measures foreseen to prevent damage to the fuel.
- After the occurrence of damage to the fuel in the reactor core,
- After the failure of the reactor pressure vessel or the pressure tubes.

VIII.1.2.6.2 Describe the accident management measures and the plant design features to protect the containment integrity after the fuel damage occurrence:

- Prevention of hydrogen deflagration or hydrogen detonation (inerting, recombiners, or igniters), taking into account the capacity of the contention venting processes;
- Prevention of the containment over-pressurization. If a radioactive material release to the environment to protect the containment were needed, it should be assessed whether this release could be filtered or not. In this case, the available means to estimate the amount of radioactive material released into the environment should also be described;
- Prevention of re-criticality;
- Prevention of basemat melt through (containment flooding at different levels to prevent the reactor pressure vessel failure or to limit the interaction of the molten core with the concrete).
- Need of supply of electrical AC power, DC power and compressed gas to the necessary equipment to protect the containment integrity.

VIII.1.2.6.3. Describe the accident management measures currently in place to face the various stages of a scenario of loss of cooling function in the fuel storage pools:

- Before and after losing adequate shielding against radiation (loss of water column shielding)
- Before and after occurrence of uncovering / exposure of the top of the fuel.
- Before and after occurrence of the fuel degradation (fast cladding oxidation with hydrogen production)

VIII.1.2.6.4. Additional aspects:

For VIII.1.2.6.1, VIII.1.2.6.2 and VIII.1.2.6.3 items, if applicable, it should:

- Identify any cliff-edge-effect that might occur and evaluate the available time before it occurs;
- Assess the adequacy of the existing management measures, including guides and procedures, to cope with a severe accident, and evaluate the potential for additional measures. In particular, the following should be considered:
 - o The suitability and availability of the required instrumentation;
 - o The accessibility and habitability of the plant's essential areas (main and secondary control rooms, emergency response centres, local controls and sampling points, repair possibilities etc.).
 - o Potential hydrogen accumulations in other buildings than that of containment.

The following aspects should also be considered:

- Organization to adequately manage the situation, including:
 - o staffing, resources and shift management;
 - o use of off-site technical support for accident management and a place from where the accident is managed, including contingencies if it becomes unavailable);
 - o Procedures, training and exercises;

- Possibility to use existing equipment;
- Provisions to use mobile devices. Availability of such devices, appropriate connectors for coupling, time required for them to be available on-site and in operation, and the site accessibility.
- Provisions for and management of supplies (fuel for diesel generators, water, etc.)
- Management of possible radioactive releases and provisions to limit them;
- Management of workers' doses and provisions to limit them;
- Communication and information systems (internal and external);
- Foreseen long-term activities (post-accident).

The envisaged accident management measures shall be evaluated considering that the situation could cause:

- Extensive destruction of the existing infrastructure around the plant, including the communication facilities (this would make the technical and personnel support from outside more difficult);
- Impairment of work performance, including the impact on the accessibility and habitability of the main and secondary control rooms and the plant emergency centre, due to high dose rates, radioactive contamination and the possible destruction of some on-site facilities
- The need to analyse the feasibility and effectiveness of accident management measures under the conditions of external hazards (earthquakes, floods etc.);
- Unavailability of power supply;
- Potential failure of instrumentation;
- Potential effects from other neighbouring plants at site.

Besides, the conditions that would prevent or make difficult for the staff, to work in the main or secondary control room, as well as in the plant emergency/crisis centre should be identified, as well as the measures that could be taken to avoid the occurrence of these conditions.

VIII.1.2.7. ON-SITE EMERGENCY MANAGEMENT

In order to meet the requirements for this assessment, it is required to report the status of the issues considered in the following items, as well as any measures deemed necessary to be taken.

VIII.1.2.7.1. Management and Control:

Verify the existence of:

- a) The procedures in place to relocate in safe areas the personnel that should remain in the plant (including operating personnel)
- b) The procedures related to agreements with external organizations and their activation protocols.
- c) Plans for evacuation and personnel assistance.
- d) The procedures for preparation and coordination for the reception of materials, equipment and additional personnel, in particular operating personnel from other plants.
- e) The necessary professional personnel resources, taking into account the possible loss of operating personnel, making it essential to have a minimum number of qualified persons to perform these tasks.
- f) Identification of external organizations and facilities that might have the knowledge and resources to help manage the emergency. For this case, the relevant protocols/agreements should be included to ensure this aid, if it is required, from the start of the accident until the ARN takes charge of the emergency management.
- g) The procedure should consider the possibility of sharing personnel in places with more than one unit, as well as the resources required for the accident management in all units of the site.
- h) The safe areas to store materials and equipment needed to implement the emergency response plan. The most important aspects to consider in the evaluation include the location of these facilities and the availability of both the equipment and the necessary means.
- i) The establishment of safe areas protected from potential fires and radiation exposure.

- j) The establishment of appropriate areas free of obstacles, to facilitate the external assistance arrival of light air vehicles (helicopters).
- k) The procedure and the means to ensure the effective guarantee of the communication among the emergency management, the control room, the safe areas, the operation shift, the recovery team, the fire fighting equipment, other external organizations, ARN, etc., as well as the communication among the different groups acting throughout the mitigation process (fire-fighters, recovery personnel, operation personnel, external organizations, etc.). In particular matters relating to education and training of such personnel should be considered.
- l) The assessment of the availability and compatibility of the necessary communications equipment, as well as the need of additional emergency equipment, batteries and chargers for this equipment, etc.
- m) The availability of sufficient portable fire extinguishers, ventilation equipment, breathing apparatus, personal protective equipment, etc., conveniently located in diverse and accessible places.
- n) The planning and availability of appropriate emergency lighting equipment to facilitate performing the activities inside and outside the different plant buildings. The existing capacity to feed and recharge such equipment, so as to ensure their operation for long periods, must be analyzed.
- o) The availability of cables, prefabricated connections, adapters, hoses, etc., necessary for the cooling and venting systems operation, as well as to ensure the instrumentation operability and the critical elements required according to the mitigation strategy.
- p) The availability, in the appropriate locations, of the applicable plans and procedures.
- q) The analysis will take into account aspects related with the personnel radiation protection as well as the use of dispersion models suitable for calculating the involved doses and reducing them to the minimum. The possibility of estimating doses in areas where intervention is necessary will be analyzed.

VIII.1.2.7.2. Fuel Damage Mitigation

The analysis of the strategies to maintain or restore the effective fuel cooling of both the core and the spent fuel storage pool, as well as to maintain the radioactive material confinement function, taking into account the personnel resources needed to deal with recovery actions, the accessibility and the means to ensure communication among the different acting groups, the procedures and the necessary equipment to develop the planned strategies and the suitable training of both the plant personnel and the external support personnel, should be considered.

The fuel damage mitigation strategy should consider:

- a) The review and assessment of the scope of the guidelines for severe accident management.
- b) The procedure for filling tanks or water storage reservoirs used in the plant cooling systems (e.g.: water supply tank, condensate tank, etc.).
- c) The procedures for manual operation of systems and components, including the use of motor/turbine driven pumps, etc., provided for the case of absence of the corresponding electrical power supply.

For the fuel stored in pools, the water cooling should be optimized to delay fuel damage should the pools be emptied by adopting measures that consider the following criteria:

- d) Availability of an alternative supply of emergency cooling water.
- e) Leakages reduction and recovery action implementation.
- f) Measures to face adverse radiological conditions in the fuel storage pool building.

Additionally, as a long-term measure, the possibility should be considered of:

- g) Providing natural and forced air circulation:
- h) Optimizing the spent fuel arrangement in its housing to distribute the heat load adequately.

VIII.1.2.7.3. Reduction of Radioactive Emissions

In case the actions included in the severe accident management guidelines and the special fuel damage mitigation measures are not successful, the release of radioactive material to the outside should be minimized. For this, measures should be taken prior to the start of such a release,

considering that once it begins, the surrounding areas may be inaccessible due to the existing radiation levels. For this purpose, the following should be performed:

- a) Analyse the availability of a suitable water supply to be used in the most convenient way.
- b) Detail the ability to use the fire fighting equipment (water and foam) for washing of possible radioactive emissions.
- c) In the case of washing during an extended time or any other actions that could lead to a significant quantity of liquid spills, the availability of temporary deposits to store large volumes of radioactive water in a controlled way, sand bags to facilitate temporary dike construction or stabilizing materials to control and retain the flow of the water used, etc., should be analyzed
- d) Consider the possibility of using compounds that retain or absorb any radioactive particles present, in order to facilitate subsequent cleaning and decontamination tasks as well as aerosol depurations.

For each of the above, their relationship with the following issues (listed in VIII.1.2.7.4., 7.5. and 7.6.) should be indicated:

VIII.1.2.7.4. Revision of Procedures

It should:

- Reassess the scope of the existing procedures applicable to emergency management strategy, including fire fighting and the corresponding recovery actions.
- Analyse the interferences between the security components and the necessary mobility and ease of access during emergencies.

VIII.1.2.7.5. Personnel Training

Suitable education and training for both the plant personnel and the external support personnel is a basic element of the mitigation strategy. In this regard NA-SA must verify that it has:

- a) The program of exercises and practices.
- b) The program of familiarization with the plant.
- c) Coordinated exercises among external organizations and personnel from operation, maintenance, recovery, etc.
- d) The joint training with external organizations (assistance protocols).
- e) The operating procedure to use both special equipment and materials.
- f) The training program for shared personnel in sites with more than one unit.
- g) If applicable, the training program of the personnel belonging to external organizations to the plant or to other similar plants.
- h) The training program for the equipment and instrument connection to alternative electrical power and water supplies, as well as the training program in the operation of devices under critical or degraded situations.
- i) The training program in the use of devices, accessories and special clothing.

VIII.1.2.7.6. Equipment

Information must be given on:

- a) The availability of appropriate connectors used in the required auxiliary equipment (e.g.: alternative supplies for electricity, water or air), as well as the procedures necessary for their effective alignment.
- b) The capacity of both the required electrical power and the cooling to maintain the plant's safety functions, as well as the availability of the easy interconnection of the equipment and the aligned components to the different electrical power supplies and to the auxiliary cooling loops and the power for the instrumentation and other devices required.
- c) The possible impact on other key plant areas (control room, switch room, cables, relays, breakers, etc.) that can cause the loss of electrical power.
- d) The availability of auxiliary equipment used for recovery actions in diverse and safe areas.
- e) The existence of an appropriate maintenance, testing and inspection program for all equipment to be used in the mitigation strategy to.

CONVENTION ON NUCLEAR SAFETY

