República Argentina

ARGENTINEAN NATIONAL REPORT FOR THE CONVENTION ON NUCLEAR SAFETY

FOURTH REPORT

September 2007

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. (NASA) and Comisión Nacional de Energía Atómica (CNEA).

Argentinean National Report for the Convention on Nuclear Safety - Fourth Report

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GLOSSARY

ABACC	*Agencia Brasileño-Argentina de Contabilidad y Control de Materiales Nucleares* (Argentine-Brazilian Agency for Accounting and Control of Nuclear Material)
AECL	Atomic Energy of Canada Limited
AMP	Ageing Management Program
AREVA	AREVA NP
ARN	*Autoridad Regulatoria Nuclear* (Nuclear Regulatory Authority)
BW	Babcock & Wilcox
CAREM	CAREM reactor prototype
CDF	Core Damage Frequency
CIAS	*Comité Interno Asesor de Seguridad* (Internal Safety Advisory Commitee)
CNA I	*Central Nuclear Atucha I* (Atucha I Nuclear Power Plant)
CNA II	*Central Nuclear Atucha II* (Atucha II Nuclear Power Plant)
CNE	*Central Nuclear Embalse* (Embalse Nuclear Power Plant)
CNEA	*Comisión Nacional de Energía Atómica* (National Atomic Energy Commission)
COG	CANDU Owners Group
CONUAR	Combustibles Nucleares Argentinos S.A.
CRT	*Comité de Revisión Técnica* (Technical Revision Commitee)
ECCS	Emergency Core Cooling System
ENACE	Empresa Nacional de Energía S.A.
ENSI	Empresa Neuquina de Servicios de Ingeniería S.E.
EOL	End Of Life
GIS	Geographical Information System
GRS	Gelleschaft für Anlagen und Reaktorsicherheit
IAEA	Internacional Atomic Energy Agency
ICRP	International Commission on Radiological Protection
INES	International Nuclear Event Scale
INSAG	International Nuclear Safety Group
IPERS	International Peer Review Service
IRS	Incident Reporting System
ISI	In Service Inspection

KWU	Kraftwerk Union
LOCA	Loss of Coolant Accident
NASA	Nucleoeléctrica Argentina S.A. (Licensee)
NERS	ARN-Nuclear Emergency Response System
NPP	Nuclear Power Plant
OPEX	Operating Experience
OSART	Operational Safety Review Team
PHWR	Pressurized Heavy Water Reactor
PLEX	Plant Life Extension
PLIM	Plant Life Management
PSA	Probabilistic Safety Assessment
PSR	Periodic Safety Review
QA	Quality Assurance
QMS	Quality Management System
RPV	Reactor Pressure Vessel
SAMP	Severe Accident Management Program
SAR	Safety Analysis Report
SIEMENS	SIEMENS Kraftwerk Union AG
SC	Safety Culture
SCK/CEN	Studiecentrum voor Kernenergie
SMA	Seismic Margin Assessment
SSC	Structures, Systems and Components
TÜV	*Technischer Üeberwachungs Verein, Baden* (German Inspection Organisation)
UNIPI	University of Pisa
UNSCEAR	United Nations Scientific Committee on the Effects of Atomic Radiation
USNRC	US Nuclear Regulatory Commission
UTN-BA	*Universidad Tecnológica Nacional - Buenos Aires* (National Technological University)
WANO	World Association of Nuclear Operators

INTRODUCTION

This Fourth 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. The present report also takes into account the observations and discussions made during the third review meeting. The conclusions made in the first review meeting about the compliance by Argentina with the obligations of the Convention are included as Annex I, those belonging to the second review meeting are included as Annex II and those belonging to the third review meeting are included as Annex III. The questions and answers originated at the third review meeting are included as Annex IV.

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

With this same purpose, Annex V shows some design characteristics of NPPs in operation Atucha I (CNA I) and Embalse (CNE); Annex VI describes the design characteristics of the Atucha II NPP under construction (CNA II), and Annex VII shows information relative to the CAREM reactor Prototype (CAREM) whose construction is anticipated in the Argentine nuclear program.

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 No 24776, approving the Convention adopted on September 20th, 1994. According to the provisions in Section 5 of the Convention, each Contracting Party shall submit for its examination a National Nuclear Safety Report containing the measures adopted to comply with the corresponding obligations.

This Fourth Report describes the actions the Argentine Republic has carried out since the third nuclear safety report was issued (March 2004) until 2007, 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.

The country has two NPPs in operation, CNA I and CNE, which initiated their commercial operation in 1974 and 1984 respectively. Their corresponding net electric power are 335 MW and 600 MW, which represents some 4.1% of the installed electric power. Both NPPs supply about 7.1 % of the total electric power generated (2006 value).

CNA I is located about 100 km to the Northwest of Buenos Aires City. The reactor is of PHWR type with a pressure vessel. 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 the pressure tube type, loaded with natural uranium fuel and moderated and cooled with heavy water.

Atucha II NPP (CNA-II) is under construction, very near CNA I. Its reactor will also be of the PHWR type with a pressure vessel, loaded with natural uranium fuel, moderated and cooled with heavy water, with a net electric power of 693 MW.

The CAREM reactor prototype (CAREM) is an Argentinean design in a development stage, with a planned electric power of 27 MW, and will probably be constructed near CNA I and CNA II NPPs. Its principal design characteristics are:

- *Integrated primary system.*
- *Self-pressurization.*
- *Safety passive systems.*

Additionally, a feasibility study has begun to define the type of reactor and the appropriate siting for a fourth NPP.

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 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 24804, 1997 or "National Law of the Nuclear Activity" is the legal framework for the peaceful uses of nuclear energy. Article 1 of Act No 24804, states that concerning nuclear matters the State will establish the policy and perform the functions of research and development and of regulation and control, through the National Atomic Energy Commission (CNEA) and the Regulatory Body (ARN) respectively.

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.

In the case of nuclear power generation, Article 4 of Decree 1540, 1994 sets that a State Company named Nucleoeléctrica Argentina S.A. (NASA) is in charge of the operation of CNA I and CNE and of the construction, commissioning and operation of CNA II.

1.3. NATIONAL PROGRAM CORRESPONDING TO NUCLEAR INSTALLATIONS

The National Executive Power authority announced important decisions in relation to the Argentine nuclear program:

- *The activities of the completion of the construction of CNA II have been remarkably accelerated with the aim of its commissioning and operation*
- *The construction and commissioning of the CAREM (Decree of the National Executive Power 1107/2006) 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 joint responsibility of CNEA and Emprendimientos Energéticos Binacionales S.A. (EBISA).*

As a consequence of this renewed impulse in the Argentina nuclear industry, NASA and CNEA signed in the year 2006 a tripartite global agreement with Atomic Energy of Canada Limited (AECL) that includes the CNE life extension and a feasibility studies for a fourth NPP, as a first step of the ambitious program of activities that are to be developed and undertaken by NASA and CNEA for the concretion of both projects.

Concerning the NPPs in operation, safety assessments carried out by the ARN from the foundation of CNA I and CNE, of which the most relevant results are detailed in this and previous Reports, indicate that no objection is found for their continuing commercial operation in compliance with the Argentine regulatory standards in force in the country, and the relevant international standards for the nuclear industry.

1.4. SUMMARY OF THE MAIN SUBJECTS CONTAINED IN THE REPORT

The present National 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, and the most significant conclusions introduced during the third review meeting in 2005. This means that the Report has been ordered 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 which are part of the Report and its complementary Annexes show the compliance of the Argentine Republic, as a contracting party of such 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 3rd Review Meeting. Chapter 3 includes detailed material that demonstrates how Argentina 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 Argentinean 1st, 2nd and 3rd 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 organisation 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 concerns functions, responsibilities and organisational structure of the Regulatory Body, human and economic resources, personnel qualification and their relationships with other official and private organisations.

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 to 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 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 describes 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 organisations, with special emphasis in training exercises on the emergency plan application.

Article 17 summarises the studies related to NPPs siting and site re-evaluation studies.

Article 18 analyses the design and construction of NPPs and their compliance with the Argentine standards as well as application of good international practices principles of defence in depth, diversity and redundancy.

Article 19 analyses 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, 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 harmonization of ARN standards with IAEA safety documents, development of new standards, quality management system in the Regulatory Body, training of human resources in the Regulatory Body, new tools and issues of safety assessment, risk-informed approach to decision-making in the regulatory body, and improvements in the emergency preparedness.

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

1.5. ANNEXES

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

***Annex I** presents the conclusions made in the first review meeting about the compliance by Argentina with the obligations of the Convention.*

***Annex II** presents the conclusions made in the second review meeting about the compliance by Argentina with the obligations of the Convention.*

***Annex IIII** presents the conclusions made in the third review meeting about the compliance by Argentina with the obligations of the Convention.*

Annex IV *includes the questions and answers discussed at the third review meeting.*

Annex V *describes the technical characteristics of the NPPS in operation: CNA I and CNE.*

Annex VI *describes the technical characteristics of the NPP in construction: CNA II.*

Annex VII *describes the main technical characteristics of CAREM reactor Prototype*

Annex VIII *presents examples of lessons learned and Corrective Actions resulting from National and International Events and Operating Experience.*

Annex IX *resumes the Quality Assurance Program of the Licensee.*

CHAPTER 2

FOLLOW-UP FROM THE THIRD REVIEW MEETING

At the third Review Meeting in 2005, several countries raised issues and made recommendations for Argentina to follow up on this fourth reporting of specific topics. These topics are discussed in the following sections.

2.1. SEVERE ACCIDENT MANAGEMENT PROGRAMME IMPLEMENTATION.

In September 2003 the ARN required to the Licensee the development of a Severe Accident Management Program (SAMP) for CNA I and CNE, starting with CNA I.

The activities covered during 2003-2004 in SAMP for CNA I (and communicated in the third nuclear convention), were:

- Development of internal procedures for the program (Organization, QA , etc).
- Selection of a methodology to characterize plant damage scenarios.
- Selection of six plant damage state scenarios to be characterized during the first stage of the program.
- Start of the development of a model for severe accident progression. MELCOR package was selected.

The activities covered during 2004-2006 in SAMP for CNA I are those related to plant damage states characterization, identification of new scenarios, grouping of accident sequences, proposal of strategies, issuing of reports corresponding to findings coming from better knowledge of plant behaviour, as well as improving what was used in previous PSA analysis. It could be mentioned:

1. *Plant damage states*

   This task includes the development of new deterministic studies (or extending the existing ones) with RELAP code up to the initiation of core uncoverage.

2. *Strategies proposal (preventive strategies for the time).*

   - *Small LOCA with failure of low pressure emergency water injection.*
   - *Small LOCA plus moderator pumps failure which delays low pressure emergency water injection because of depressurization delay.*
     - *This is possible only for a particular break size.*
     - *The strategy proposed uses the volume control system pumps to feed water into the primary. At the same time they are fed through a pump that takes water from a demineralized light water reservoir.*
     - *The strategy proved to be effective as a preventive measure for breaks up to 20 cm^2 in any primary circuit location.*
   - *Blackout*
     - *The strategy comprises two manual actions. One is to avoid air entering the primary, which deteriorates Secondary Heat Sink (SHS) action (through SGs). The other is to avoid, or at least significantly reduce, the probability of LOCA scenarios.*
     - *A Shift Supervisor is working with the SAMP group in order to identify the design changes necessary to incorporate the strategy.*
   - *Unavailability of the second heat sink pumps or lack of sufficient inventory for different accidental scenarios.*
     - *The strategy applies to all scenarios in which SGs are the available heat sink and SHS (emergency water supply system) tank or pumps are partially or totally unavailable. The objective is to extend SHS mission time by adding water to the SHS tank, or directly to the SGs, in order to have more time to recover the plant safety equipment necessary to keep the plant in a controlled shutdown. The strategy takes into consideration equipment that is already in the plant.*

At present a new strategy is analyzed related to the use of an alternative control room in case the main control room becomes uninhabitable. The postulated scenario is fire in the main control room.

It is also under analysis the possibility of venting the containment considering different alternatives of opening the sphere, taking into account the experience in other German design containments.

3. *Progress in the development of the accident progression model*

   The work was focused mainly on the revision of the previous model in order to improve it; the revision of input plant data and the development of specific models for reactor coolant system, moderator, pressurizer and steam generators. A single channel model was also developed in order to better analyze the behavior of the fuel element and its channel during core damage.

   The results of these analysis were presented at the meetings Cooperative Severe Accident Research Program (CSARP) and at the MELCOR Users where the peculiarities of CNA I were discussed with other specialists that attended the conference.

4. *Other issues related with severe accidents*

   Some years ago, the Regulatory Body started to study different issues related with severe accidents such as containment failure modes of CNA I, hydrogen behavior and associated mitigation systems, with the purpose to establish the basis for potential requirements. Currently, as a new model of severe accident progression is being developed with MELCOR for CNA I, these former studies are no longer used.

   Besides, as it was pointed out in the third Report, the design of CNA II considers hydrogen mitigation devices. The design of these devices is currently being evaluated. The experience gained through these devices design will be used as a basis for CNA I evaluations.

5. *Severe Accident Management team definition*

   With the collaboration of CNA I plant personnel the Licensee is defining the insertion of the SAM team in the Internal Committee of Emergency Control structure. Besides, 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.

   In connection with CNE accident progression analysis, the study carried out for "Generic CANDU Probabilistic Safety Assessment" and the "CANDU 6 Probabilistic Safety Study" developed by AECL is considered, and a number of core damage accidental sequences were selected as results from the PSA level 1 developed. On the other hand, the Licensee is maintaining close discussions with AECL to develop an international methodology for analysis of accident progression in CANDU reactors.

More information about this subject is given in Subsection 3.14.3.1.2 of this report.

2.2. APPLICATION OF PROBABILISTIC SAFETY ASSESSMENT RESULTS

The activities covered during 2004-2006 in the applications of PSA are the following:

2.2.1. CNA I PROBABILISTIC SAFETY ASSESSMENT APLICATION

The outcomes of the PSA study for CNA I has been developed to an enhanced level 1 according to the subjects related with the back-fitting program. That program had added new safety systems, such as the Second Heat Sink and redundancy improvements as in the secondary relief valve system.

The enhanced PSA is undergoing a revision by the Licensee on the internal events subject. This matter is aimed at a Core Damage Frequency (CDF) result more realistic than the latter, by adding the outcomes of more recent thermo-hydraulic studies that are being performed in the frame of Severe Accident Management Program. Conservative hypothesis adopted in former studies can be ruled out with new ones in force. E.g., small LOCA through the pressurizer relief valve along with the Low Pressure Safety Injection failure, use to be the accident with the largest contribution to the CDF. Such accident sequence does not lead to a core damage during the mission time, according to the most recent studies.

In February 2005, NASA submitted to ARN a technical evaluation of the impact on the nuclear safety, to change the period between planned outages, from 12 to 18 months. NASA evaluated this impact when frequencies of preventive maintenance, routine tests and in service inspection programs are to *be modified. The augmented test period was the main subject considered and an estimation of the impact on the CDF using PSA was carried out, which turned out to be $1,3x10^{-5}$ /year.*

A specific acceptance criterion for a CDF increment due to design changes is not currently included in the Argentinean standards. The US-NRC recommendation on this subject sets a limit to the increased value of CDF due to design modifications. This limit is set to $1.0x10^{-5}$/year.

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 changes.

At present an update of CNA I PSA is ongoing, taking into account new findings coming from a better knowledge of the plant behavior as was mentioned before, and CDF frequency is expected to be reduced.

2.2.2. CNE PROBABILISTIC SAFETY ASSESSMENT APLICATION

A PSA with corrective actions was accomplished by the Licensee since 2004 until now. The ARN asked for such actions to be included within the operation procedures and the re-training program of the personnel.

The scope of the PSA spanned to the low power state. That yielded an $8.8x10^{-6}$/year, figure which is 35% of the full power one.

A qualitative study on other potential sources of radioactive release (spent fuel elements bay, Cobalt rods, spent fuel elements dry storage, etc) was also carried out by NASA as an improvement of the study.

Regarding other PSA applications, it can be mentioned that the impact of the "mejillón dorado" bivalves, which clogs the heat exchanger and reduces its efficiency, was evaluated. NASA as a requirement of the ARN did this study. It was based basically upon considering some affected cooling system, such as the Service Water System and Emergency Water Supply System. It was found that the ECCS was the most impaired system, since it has only one heat exchanger that exchanges heat to the secondary side. This component was tagged as "non repairable" and a service inspection document was issued: a bi-monthly surveillance flow measurement. Such criterion was applied to other heat exchangers, cooled by service water. A surveillance of those components is also part of the preventive maintenance program. Analyzing the expected changes in the CDF figure was adopted as a main criterion in this case.

Another application of the PSA was the extended period between outages from 12 to 18 months. The ARN requirement, as in CNA I, was to evaluate the impact on the nuclear safety when frequencies of preventive maintenance, routine tests and in service inspection programs are to be modified. As a part of the requirement, NASA developed a study using the PSA level 1 tool with the plant at full power. Such study met the criteria established by the US-NRC and adopted by ARN to limit the increased value of CDF due to design changes to $1.0x10^{-5}$/year. The ARN authorized the increased period between outages from 12 to 18 months with a three months tolerance margin.

More information about this subject is given in subsection 3.14.3.2 of this report.

2.3. CNE SPECIFIC SEISMIC PSA

CNE is located in a seismic risk area. PSA was not considered in the original design but many specific requirements were made related to seismic risk:

- *Plant responses in case of an earthquake occurrence,*
- *Seismic instrumentation verification,*
- *Fuel Elements load machine seismic responses verification,*
- *Services buildings response in case of an earthquake occurrence.*

These requirements were met by NASA at the right time.

Later on, ARN required the application of the PSA methodology in CNE. The scope included the seismic PSA.

In order to comply with this requirement, that involves the safety re-evaluation in case of a seismic occurrence at CNE, the Licensee proposed to ARN the application of the Seismic Margin Assessment (SMA) methodology instead of the seismic PSA. That proposal specified the use of the EPRI methodology.

It could be mentioned with respect to this issue, that the internationally accepted and more used methodologies for seismic risk evaluation are, the seismic PSA and the SMA methodology. The *international experience allows to verify that the conclusions obtained by both methodologies are equivalent as far as the implementation of improvements to NPPs is concerned. However, the time and resources required to perform the re-evaluation are considerable lower in the case of SMA, in comparison with seismic PSA.*

ARN agreed with the NASA proposal with the following conditions:

- *To justify the value to be used for the Review Level Earthquake (RLE) for the CNE site, the intensity and floor spectrum, and the attenuation factors to be used.*
- *Taking into account that EPRI proposes three different scopes, ARN required the use of the "total reach" EPRI methodology scope.*

The SMA study began to be performed at CNE applying the methodology developed by EPRI. During the recent planned outage in 2007, a plant walk-down made by specialists was performed that involved fundamentally the SSCs which are inaccessible during the operation at full power. The evaluations of the results of that walk-down are in progress.

Cordoba University is also updating the seismic data referred to the siting and the spectra in the different levels of the building. Some of the specialists of this institution took part in the seismic studies performed before the CNE commissioning at the beginning of the eighties.

It is important to point out that NASA communicated to ARN its intention of extending CNE lifetime and the safety assessment on seismic occurrence is part of the basic studies to define and authorize the lifetime extension. The studies are expected to be finished in time for the next Nuclear Safety Convention.

More information about this subject is given in Subsection 3.17.6.3 of this report.

2.4. QUALITY MANAGEMENT SYSTEM IN THE REGULATORY BODY. CONTINUING IMPLEMENTATION

The ARN, as a "de facto" and "de jure" independent authority, has always taken and takes actions with the objective of improving the quality of its regulatory performance.

In 2002, the ARN Board of Directors decided to strength the quality management activities by planning its development and implementation. The support of external advisors involving the National Technological University of Buenos Aires (UTN-BA) was procured to achieve a well-structured quality management system, focused on external and internal stakeholders and based on the "Continual Improvement Approach". This goal was achieved in 2005.

For an appropriate implementation of the Quality Management System (QMS), ARN created in 2005 the Quality Management Unit, which depends on the Board of Directors.

ARN has established, documented and implemented the QMS according to the requirements established in the ISO 9001:2000 Standard. The actions and requirements are described in the Quality Manual (MC-ARN), where the ARN Board of Directors declares and communicates the Quality Policy and Commitment.

Additionally the Board of Directors has decided to initiate the discussion, interpretation and implementation of the IAEA Safety Requirements GS-R-3-Management System for Facilities and Activities.

The QMS in the ARN is implemented by applying a processes approach. Seven regulatory processes (nuclear installations, radioactive installations, transport, radioactive waste management, prevention of intentional events, special projects and non-proliferation) and four support processes (institutional communication, human resources, research & development and financial and material resources) have been identified.

The sequence and interaction of these processes are established and represented in charts. Each process is described in a Process Letter where the objectives, inputs, outputs, checkpoints, associated documents, performance indicators, non-conformances and corrective actions are considered.

The analysis and measurements of these processes are carried out by self-assessments, and Effectiveness and Efficiency Indicators are been defined.

ARN carries out quality self-assessment in order to:

- *demonstrate the products and processes compliance with stakeholders 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.*

The self-assessments are performed by qualified personnel. They are independent of the area which is assessed. During 2006 and until June 2007, eleven (11) assessments were carried out; 17 non-conformances, 10 observations and 63 opportunities for improvement were identified.

The QMS is based on a solid documental structure, which involves a Quality Manual, General Procedures, Specific Procedures, Working Instructions, Forms and Records.

Up to June 2007 there were ninety-one (91) approved documents and twenty-seven (27) documents are being prepared at present. Documents and records control and technological security controls of information have been implemented.

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

ARN has implemented communication mechanisms concerning:

- *legislation, standards, regulatory guides, licenses, permits, authorisations, and*
- *feedback from stakeholders regarding consultations, complaints, opinions and opportunities for improvement.*

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

*In ARN Website - **www.arn.gov.ar** -, regulatory standards, laws and acts, permits and Operating Licences information, annual reports and general public communications among others can be found.*

Concerning the QMS Planning, the ARN Board of Directors approves annually the Work and Budget Plan, which contains the 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, so as to assure its continuity, effectiveness and efficiency.

A new aspect related with human resources, and that can be considered included in the QMS is the Knowledge Management. In early 2006, the ARN started an initiative on regulatory Knowledge Management to give consideration to the generation gap, the loss of knowledge resulting from retiring experts, and the need to transfer such knowledge to younger generations through training. As a result, two projects have been initiated so far.

First, to find and turn both tacit and implicit knowledge explicit. For this reason an in-depth interviewing approach was used with ARN's experts. Eighteen experts have been interviewed so far and most of the tape-recorded interviews have been transcribed. The results of these interviews have been used to find possible projects between the experts and the new workers and to strengthen the training programs in the organization. The methodology used was the "History of the Learning Process".

Second, with the same objective in 2007 a mapping process has been initiated on one regulatory knowledge domain of the organization.

2.5. CNA I – REACTOR PRESSURE VESSEL SAFETY ASSESSMENT

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

Initially, KWU (the Designer), didn't consider necessary to formulate a surveillance program for the RPV's material because it estimated a very low fast neutrons fluence (E>1 MeV) in the beltline region, for it to produce important changes in the material brittle to ductile transition temperature throughout the CNA I design life time (EOL- end of life, corresponding to 32 years of full power operation). Nevertheless, during further evaluations the neutron fluence was found to be able to reach higher values, and 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.

The program consisted in irradiating samples of the RPV's most critical base material as well as their respective welding, in the lower part of some of the CNA I cooling channels. The differences in the *neutronic spectrums in these positions respect of the RPV's wall, led to an important uncertainty in the evaluation of the results, which implied not been able to use the results in a safety study.*

The first valid irradiation tests within the irradiation program in host reactors were undertaken by KWU in 1983 at the German VAK reactor. They were done on irradiated and also non-irradiated samples and consisted of various kinds of tests: tensile, Charpy impact, drop weight and fracture toughness. Although the results showed that the RPV should operate safely until EOL, the Regulatory Body required to the Licensee additional safety studies.

Those studies consisted in performing further irradiations and testing more samples, and making new evaluations on neutronics and Pressure Thermal Shock (PTS) subjects. For those task the Licensee received the advise of KWU/Siemens, experts and international consultants.

An important milestone which should be taken into account concerning the RPV's material, was the technical meeting which took place in CNA I in the beginning of 2002, with the participation of international experts from various technical scientific institutions (Oak Ridge National Lab., GRS, VTT-Technical Research Centre of Finland, and FRAMATOME ANP) and experts from NASA, CNEA and ARN. During this meeting everything done up to that moment related to the RPV material properties was discussed and further studies were proposed, which were all undertaken.

These additional studies comprised additional irradiation programs in research reactors with neutron spectrums and irradiation temperatures similar to those of the CNA I RPV. The research reactors that were used for the irradiations were the BR2 in Belgium (2003) and the Loviisa in Finland (2003 - 2005). In those occasions the irradiated samples underwent diverse tests at internationally acknowledged laboratories.

In June 2006, NASA submitted to ARN a report done by an international consultant lead by CEN/SCK of Belgium. The report presents the complete and detailed analysis of all the data and results obtained from the CNA I RPV surveillance program, and the tests of the irradiated samples at VAK, BR2 and Loviisa, as well as presents a safety evaluation of the RPV for EOL and for 1.5 times EOL.

This report shows the re-evaluation done on the 87 data samples that contain the 770 test results using the internationally developed lasted tools and a safety evaluation. For the safety evaluation, the acceptance criteria defined in the USNRC - NRG 1.99.2 regulatory guide, were applied.

The report also gives numerous results on irradiations done on similar material to that of CNA I's RPV, which support the hypothesis of the absence of the "flux effect" or "lead factor" in the accelerated irradiations done on material from CNA I in the above mentioned research reactors.

From the conclusions it should be highlighted the demonstration of the consistency of the results of all the test undertaken in the host irradiation programs and a model that explains the uncertainties from those corresponding to the surveillance program (samples irradiated at the CNA I RPV).

The report concludes that the integrity of the CNA I's RPV is guaranteed until EOL and 1.5 EOL. Also can 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.

Also could be mentioned that in addition to the tests and evaluations done to the RPV´s material, non-destructive test were undertaken on the primary welding of the RPV and on the heat affected zone, as part of the In Service Inspection program, whose results show that so far no relevant flaws indications have been found .

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 end of life. Nevertheless, the RPV integrity assessment is considered a permanent subject, for that reason, a continuous evaluation is also necessary in the areas of non destructive examinations, PTS and neutronic, including the results from the periodical 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.

2.6. REGULATORY AUDIT TO CNA I OPERATING EXPERIENCE FEEDBACK

During 2006, according as it was planned, the ARN made an Audit to the CNA I Operating Experience Management Group. The Audit team emphasized the considerable improvement in the follow-up of the corrective actions, in the definition of time limits for its implementation, and in the increase of human resources available for this task.

Concerning the NO CONFORMITIES that were found, they were qualified as Minors, and were related to "observations" in the events records, as well as in the principal indicators report and trends.

The Audit team also made recommendations on the use of specific techniques for the root cause analysis as well as for the classification of events.

2.7. CNA II: CURRENT REGULATORY REQUIREMENTS.

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 art related to this kind of installation, could affect adversely the installation licensing process.

ARN analysed the conditions and came to the conclusion that:

1. *There exist no restraints to continue the licensing process of CNA II, as far as the Licensee fulfils with the legal regulatory frame in force, that include specific requirements that arise from safety evaluations and inspections that will be performed in the future.*
2. *Shall the Operating License be granted, the Licensee must also fulfil with the legal regulatory frame according to Law 24.804, its Regulatory Decree 1390/98, the regulatory standards, regulatory requirements and Sanctions Regulations of ARN, and with the international legal instruments bonds 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 safeguard.*

ARN based its opinion on the following issues:

1. *CNA I is a NPP of second generation whose design dates from ´60 and has been in operation since its commissioning in 1974 with a satisfactory safety level. The installation's operation for more than three decades, the performance follow-up of other similar installations, which allows the acquisition of great operative experience that helped, in turn, to implement significant safety back fitting improvements. Among the most relevant aspects are: a new core emergency cooling system through steam generators (second heat sink), the installation of insulating valves in the pressurized auxiliary line, the safety assessment of the RPV integrity, the new design of the moderator pool's internal components and the elimination of the reactor internal "stellite-6".*
2. *CNA II has more advanced safety aspects than its predecessor CNA I, coming from the "Konvoi" design concept that was used. In fact, among other things it can be mentioned the redundancy of "2 of 4"relevant safety systems such as: the moderator and the essential electric supply, better base material of the reactor pressure vessel (with low copper and phosphorus content), easiness for early detection of fissures and the existence in the basic design of an emergency cooling system of the core through steam generators that function as second heat sink.*
3. *Apart from these original safety design aspects, the operative experience of CNA I and the international experience applicable were taken into account, such as the reactor's internal design (principally improvements in the fuel channels, in the control rod guide channels and instrumentation guide tubes), and the elimination of "stellite-6" in the core materials.*
4. *CNA II is the first NPP to whose licensing the AR 3.1.3 regulatory design standard is been applied. In CNA I and CNE this standard was not applied because it was issued after the licensing of these installations took place.*

ARN personnel with the advise of domestic and foreign institutions is carrying out the regulatory tasks of evaluation, inspection and audits. The advising is performed – as ordinary practice by ARN and of through contracts or specific agreements. In fact, contracts have been made with institutions at USA, Germany and Canada that are mentioned in Chapter 3.18.

2.7.1. CNA II CRITICAL TECHNICAL ISSUES TO BE SOLVED

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, the ARN established as first priority a set of critical technical issues to be solved. The Licensee was required to provide adequate solutions for them. The critical technical issues are:

- *Update of the Quality System,*
- *Qualification of the Design Authority,*
- *Review of Licensing Basic Criteria,*
- *Review of Safety Issues.*

Towards that end the initial action taken by ARN was to update the process required to qualify the Design Authority to be proposed by NASA in the framework of the requisites indicated in Guides IAEA NS-R-1 (design), IAEA NS-R-2 (operation) and INSAG-19.

The information given at present by the Licensee in each critical technical issue are under evaluation process by the ARN, and are shown in the following subsections.

2.7.1.1. Update of the Quality System

In order to comply with this requirement, in March 2007, NASA sent to ARN the upgrade of the Quality Management System. The new version was performed taking into account the QA original program, IAEA Practice Code 50-C-Q, ISO Standard 9001:2000 and Regulatory Standard: AR 3.6.1.

This upgrading comprises also occupational Health and Safety requirements, that were performed taking into account the 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.

2.7.1.2. Qualification of the Design Authority

With the purpose to meet this requirement and assume the project direction, during years 2004 and 2005 NASA maintained negotiations with the original designer of the plant –Siemens- in order to discuss the technical and financial conditions to resume and finalize the construction of CNA II.

According to the negotiations, NASA has the responsibility for finalizing the basic and detail engineering, the construction and the commissioning of CNA II. The CNA II detail design is 90% fulfilled. The remaining 10% will be completed by NASA with the technical assistance of different institutions. Some of them are SIEMENS, AREVA, AECL, IAEA, UNIPI, GRS, CEN/SCK and CNEA..

More details about this subject is given in Subsection 3.18.3.3.2.2 of this Report.

2.7.1.3. 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, where it is explained that the Argentine regulatory criterion does not use the concept of "maximum credible accident", and the event of Loss of Coolant (LOCA) should be taken into account in the frame of a probabilistic risk analysis

Consequently for CNA II basic design, the LOCA evaluation were carried out in the frame of the risk probabilistic analyses, out of which arose, as a design based accident, a rupture in the biggest pipe connecting the primary circuit and moderator, which is smaller than rupture 2A.

From the Probabilistic Risk Analysis applied to a big LOCA (as the 2A rupture), a value less than 10^{-7} / year frequency of occurrence was obtained. That value was obtained by extrapolation, from the respective Biblis B NPP (German) taken as reference. The adoption of that applied 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 rupture 2A of the principal pipe.

The Regulatory Body objected to the bases of the risk analysis, as it was not possible to make any judgment on the validity of extrapolating the occurrence frequency value to LOCA 2A event, defined as 10^{-4}/year for Biblis B, to a value lower than 10^{-7}/year for CNA II, as the information supplied by the designer of the risk analysis was indicative rather than demonstrative. The Regulatory Body concluded that it was necessary to require more information on the LOCA 2A probabilistic evaluation.

New probabilistic risk analysis applied to big LOCA are been developed by NASA (see next Subsection).

Besides it is worth while mentioning that a review of the present predominant basic criteria all over the world, has resulted that LOCA 2A shall be considered as an accident that must be covered by the safety systems, independently from its occurrence probability.

2.7.1.4. Review of Safety Issues

CNA II is a NPP whose original design involves that the void reactivity coefficient be positive. For that reason ARN considered that it is of high importance to analyze the installation behavior in those *events that may lead to steam or void equivalent formation in the primary circuit, where the LOCA can be considered as most relevant.*

A thorough analysis of this problem shall consider the different aspects:

a. *Compliance with the regulatory criterion defined in regulatory standard AR 3.1.3 "Radiological criteria relating to accidents in nuclear power plants".*

b. *Application of good international practices that arise from the following analysis:*

   b1. Evaluation of the Break Preclusion 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,

This complete analysis is being carried out by ARN request and connected with this it can be pointed out:

a. *Compliance with the regulatory criterion defined in regulatory standard AR 3.1.3*

   - *NASA is finishing PSA level 1 which is being reviewed by ARN.*
   - *NASA is developing PSA level 2and 3 with external counseling in case of level 2 and ARN is performing an on-line review of both PSA.*
   - *With the deterministic (see b2 below) and PSA results NASA expects to fulfil with what is established in regulatory Standard AR 3.1.3.*

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, etc,*
   - *Adequate Inspection Service Program,*
   - *Detection of leakage.*

   Many of these evaluations are being performed with external technical advising.

b2. *Calculation performed based on an adequate and validated code*

   NASA has signed an agreement with the UNIPI for the development and application of a model of CNA II utilizing Relap5/3D- Nestle (thermo-hydraulic and neutronic coupling programs) and CFX (program of computer analysis of fluid-dynamic).

   Additionally, an international "ad-hoc" expert group (IRG – International Review Group) will supervise and approve all the activities developed by UNIPI for NASA. Besides it will establish a connection between the scope of these activities:

   - *Whether the need of experimental tests that may arise as result of this analysis it is necessary or not.*
   - *The Probabilistic Safety Analysis.*
   - *The attendance to NASA due to eventual requirements or additional information request by ARN.*

   On the other hand, ARN has signed an agreement with Purdue University to carry out independent assessments, based on development and application of a model of CNA II using the programs Relap5 Mod3/Parcs (supported by US-NRC).

b3. *Improvements that can be made to the prompt extinguish design system*

   NASA has begun the study of the modification of the fast boron injection system in order to reduce the injection delays and to increase the injection velocities, as well as has analyses modifications to its basic design. NASA is now studying the present material possibilities of its modification and limits.

   NASA is also evaluating the additional introduction of negative reactivity simultaneously to boron injection.

b4. Improvements in the design in order to get a better void reactivity coefficient

NASA is analyzing 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 NASA can make 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).

2.8. CNA II: STATUS OF LONG TIME STORED EQUIPMENT.

During the time when the construction of CNA II was delayed, one of the principal concern was the appropriate components preservation.

The following criteria was used for the appropriate components preservation:

- *Influence on Nuclear Safety*
- *Economical value*
- *Replacement feasibility*
- *Impact on Project schedule*
- *Preservation cost versus Replacement*
- *Damage Sensitivity*

About component preservation tasks, they were divided in:

- *Routinely: Applicable to all components and installations. Their scopes and execution frequencies were defined in specific procedures and instructions.*
- *Non-routinely: 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 aspects were taken into account were the materials, parts and elements affected by ageing. The most important were Gaskets, Rings, Welding Electrode, 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

Also were taken into account the materials, parts and elements affected by new regulations, such as the insulation material for primary system components (experiences recently gained from sump clogging), change of refrigerant in chilled water machines (Ex.: R12 to R134a), and elements containing asbestos.

The preservation process were subjected to a continuous assessment thought out Licensee Internal and External Quality Audits, Siemens Inspections, Insurance Company Verifications and Regulatory Body Verification.

As an example the Inspections of preservation tasks performed by Siemens were the following: June and October 1986, May 1988, February and March 1989, October 1990, February and June 1991, March and December 1992, April and November 1993, March 1995, April 1996, April 1997, April 1998, October 2003 and April 2005 (Siemens/FANP walkthrough).

Additionally, in 2007, an IAEA mission took place with regards to the analysis of state of preservation of stored components and demonstration of fitness for continued use.

Personnel qualification, with the purpose of assuraring an adequate process of preservation, is one of the main concerns of NASA. In that sense the personnel that executed preservation tasks were trained and qualified according to NASA procedures, while Preservation Supervisors and Preservation Team are qualified by Siemens/FANP.

The components preservation process results could be resume as follow:

- *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 have been identified*
- *Criteria of specific revision of components and evaluation of possible replacement of parts subjected to natural ageing, will be applied during the pre-phase of the Project*

The ARN is analysing all the information submitted by NASA.

2.9. REGULATORY BODY HUMAN RESOURCES

During the period covered by this report, the ARN has increased substantially its personnel, from 202 persons at the end of 2002 to 312 at the beginning of April 2007. This increases in personnel, mainly young professionals, is due to the need to cover positions which have to be filled as the generation of specialised professionals reach retirement age, and also cover the new tasks to be undertaken by the Special Processes Unit (CNA II,.CNE life extension, CAREM reactor project Licensing and Fourth Nuclear Power Plant pre-feasibility study)

It is worth mentioning that of the 312 people working at the ARN, 80% are professional and technicians that undertake specialised technical tasks in areas of their competence or are under training programs, and 20 % perform administrative activities. Furthermore is the fact that 7% of the total personnel of the ARN hold high level positions or functions, and have a specialised training of about 20 years or more working in regulatory activities.

ARN is presently going through a period of a marked reduction of specialised personnel due to some are reaching retirement age and to the increasing offer of better-paid job opportunities from other industries.

The young professionals that do enter to ARN must undergo intensive training to attain the basic knowledge in the nuclear field. Nowadays, monetary considerations have a predominant importance over professional development in defying tasks and consequently it makes it much harder finding motivated young professionals with the desire to learn and progress in the scientific/technological sphere. In this context, the ARN is organising supportive courses for the incoming personnel and special courses for management directives.

ARN not only trains its own personnel on radiological protection and safety of radioactive and nuclear sources, it also trains technical people from other national and international institutions.

Furthermore, the ARN is using knowledge management as a useful tool to transfer and keep up the knowledge standard among its professional personnel.

Any deficiencies on the ARN response capacity due to lack of senior personnel is covered through work agreement and contracts with specialist, and domestic and foreign organisations.

More information about this subject is given in Chapter 3.8 of this report.

2.10. DEVELOPMENT OF INDIRECT SAFETY CULTURE INDICATORS

From 2006, and taking into account its own experience plus IAEA documentation, the number of safety performance indicators used in ARN was modified to 24. The set in current use is shown in Subsection 3.14.2 of this Report.

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 but not to be used alone, but together with other tools, such as, event analysis, audits, inspections, (among others), for monitoring safety.

Furthermore it should be mentioned that Argentinean experience in the use of Safety Indicators showed that they by themselves are not sufficient to assess safety during NPP operation. However they contribute to have in an estimation of the safety status and trends. Nevertheless, they represent an important tool in order to plan inspections, audits and some special regulatory assessments.

As a particular case, the development of safety culture indicators remains an open issue in the international community. In Argentina some safety performance indicators are used as an indirect tool to evaluate safety culture. They are the following:

Training:	*Number of hours devoted to training on safety-related issues*
Operating Experience:	*Number of documented event analyses, findings or design modifications in similar power plants*
Internal control:	*Number of internal technical audits*
Compliance:	*Number of pending Regulatory Requirements*
Abnormal Operation:	*Number of significant events, and*
	Impact of significant events on the core damage frequency.

2.11. PEER REVIEWS

The Licensee is a member of the WANO. Both NPPs under operation (CNA I and CNE) received WANO Peer Review Missions. Furthermore, NASA participated in all WANO programs through the WANO – PC (Paris Centre): Peer Review Missions, Technical Exchange Visits, Operative Experience Exchange Program, Lectures and Participation in Courses and Symposiums.

Besides, Argentina has provided different specialists from both NPPs to participate in 27 Peer Review Missions and Meetings organized by WANO. Detailed information about this subject is given in Subsection 3.19.8.3 of this report.

Since 2005, the following WANO activities were specifically related to CNA I and CNE:

PERIOD	***HOSTESS***	***ACTIVITY***
February 11-15, 2005	*CNA I and CNE*	*WANO Technical Support Mission (TSM) TSM–05–026–AM-RR*
June 2005	*CNA I and CNE*	*Final WANO PC Report*
December 12 -15, 2005	*CNA I*	*WANO PC Peer Review Pre visit*
June 26 - July 14, 2006.	*CNA I*	*Peer Review*
October 27, 2006	*WANO – Paris Centre*	*Exit Meeting Atucha I NPP*
Feb 26 – March 2, 2007	*CNE*	*WANO TSM on Peer Review Methodology*
May 23-27, 2007	*CNE*	*Peer Review Pre-visit*
June 19- 20, 2007;	*CNE*	*WANO TSM on Safety Culture*
June 21 -22, 2007	*CNA I*	*WANO TSM on Safety Culture*
June 25 - 29, 2007	*CNA I*	*WANO AV on Operational Experience Management*

In order to improve some particular areas in CNA I after the Peer Review, NASA requested WANO assistance through the following technical support missions:

- *Safety Culture (done)*
- *Operational experience (done),*
- *Industrial Safety*
- *Task Observations (actions in the field)*
- *Radiation Protection*
- *Follow up and support periodical visits every four months in the plant.*

Besides, WANO Peer Review follow up mission to CNA I was requested for March 2009. Also a CNE Peer Review is expected to take place on September 2007.

A WANO Technical Meeting on Self Assessment for CNA I and CNE personnel is planned for the near future.

Another peer review activity is the technical cooperation agreement that was signed between CNEA-NASA and IAEA. The objective of this agreement was an independent revision of the engineering and licensing activities of CNA II. During the first part of 2007 a mission took place regarding the analysis of state of preservation of stored components and demonstration of fitness for continued use.

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 inspections results.

3.6.2. EXISTING NUCLEAR POWER PLANTS IN ARGENTINA

Argentina has two operating NPPs, CNA I and CNE, and another under construction, CNA II.

CNA I, located some 100 km Northwest from Buenos Aires City, with a net electric power of 335 MW began its commercial operation in 1974. The reactor is of the pressure vessel PHWR type, moderated and cooled with heavy water. According to the original design CNA I was fuelled with natural uranium, but fuel elements of a new design were incorporated from 1995 to 1999, having slightly enriched uranium (0.85 % w U-235), so that the reactor core is now fully loaded with slightly enriched fuel.

CNE, which initiated its commercial operation in 1984, is located in the Province of Córdoba, some 110 km to the south of the homonymous city, and its net electric power is 600 MW. It is a PHWR reactor of CANDU type, natural uranium loaded and heavy water moderated and cooled.

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

3.6.3. NUCLEAR SECTOR ORGANISATION

In August 1994 the CNEA was divided into three independent organisations. 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 organisation is NASA (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.) an organisation acting as architect - engineer of CNA II.

The third organisation, originally named Ente Nacional Regulador Nuclear (National Board of Nuclear Regulation) and later named Autoridad Regulatoria Nuclear (ARN - Regulatory Body) by means of Act No 24804, 1997, is constituted by the regulatory branch of the former CNEA. This branch started its regulatory activities in 1958. The ARN is a completely independent organisation, 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.
- Organisation updating.
- Components inspection program.
- Periodic tests program.
- Emergency plan (see chapter 3.16).
- Training and qualification of operating personnel (see chapter 3.8).
- Quality Assurance Program (see chapter 3.13).

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 for License renewal following the IAEA recommendations.

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

3.6.4.1.2. Organisation updating

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

In the case of CNA I, the necessity of updating and improving the NPP organisation, led to some changes in the Organisational Chart(See chapter 3.13). To implement these improvements the engineering groups and those of event evaluation were reinforced. Furthermore, the Production Management was created to coordinate the tasks between the maintenance and operation sectors.

The original organisation at CNE had not required many changes up to 2005. Since then, important changes have been made to the plant engineering section, and a group has been constituted for CNE´s Life Extension (See chapter 3.13).

3.6.4.1.3. Components inspection program

These activities include the follow up of the reactor components with the purpose of detecting eventual modifications of material properties such as 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 system and component of the nuclear and conventional area, which according to applicable standards and operating experience, are considered critical for the installation safety and availability.

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

3.6.4.1.4. Periodic tests program

Both NPPs have a periodic tests program as part of the surveillance program where availability of safety and 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.

3.6.4.2. SPECIFIC IMPROVEMENT ACTIVITIES

In this section some safety improvements implemented from 2004 to 2006 are shown. For a better and orderly presentation, there are other improvement activities detailed in other chapters of the present report.

3.6.4.2.1. CNA I Replacement of Internal Reactor Components

The replacement of the fifty four 2.0 mm foil channels still remaining in the reactor was complete. Presently all the channels are of 0.4 mm zircaloy PCA-S foil and have the new design with an inferior opening / nipple.

3.6.4.2.2. CNA I Improvements in the Physical Protection System

The Physical Protection System of CNA I is presently being updated. The modification consists mainly in the installation of a double perimeter fence with electrostatic and micro-phonic sensors around all the Protected Area. A new access control system will be installed as well as a monitoring system by means of video cameras and domes. The improvement work had reached 85 % of completion by April 2007.

A new department was expressly created for this task and competent personnel were incorporated to undertake the job. Presently these personnel are undergoing specific training.

3.6.4.2.3. CNA I Improvements in the Emergency Electric System

Presently there exists an electric interconnection between CNA I and CNA II which allows the electrical supply from one of the emergency diesel generators of CNA II to supply the bus bars of CNA I. This interconnection permits a more trustworthy emergency electrical supply.

Nevertheless as the construction of CNA II has been accelerated and this current interconnection will soon be nullified, actions are being considered to maintain the present confidence level of the electric system for CNA I. A study was made in 1994 by Siemens-KWU where a design change was proposed to improve the confidence level of this system. Nevertheless, taking into account the elapsed time, contacts have been made with AREVA to revise this study taking into account the possible obsolescence of some of the components.

The proposed modification for the emergency electric system in the above mentioned study contemplates the existence of two separate grids of 6.6 kV with uninterrupted supply and also a 400 V uninterruptible mini-system configured so that safety related components, which cannot be without tension, may be connected.

The modification will allow the connection of two new Diesel generators (4 MVA each) to uncharged bars and subsequently a reconnection process will take place under 12 sec.

The 400 V uninterruptible systems will be connected to rectifiers, inverters and batteries.

The revision of this study should be completed by September 2007. This study will be taken into account for the contraction of the modification of the electric system, which must be finished by the time CNA II commissioning starts.

3.6.4.2.4. CNA I Improvements of the In Service Inspection

The following improvements must be mentioned for this period:

- *Version 3 of the In Service Inspection Manual was issued. It contemplates the incorporation of inspection areas, lowering radiological exposure by task planning and categorisation of the objectives under inspection as Conditional or Required.*
- *The removal of part of the original equipment in disuse permitted to increase the inspection of the circumference welding of the Pressure Tank in 100 % of its length using a new inspection device.*

Fulfilment of the Program: During the third inspection interval (07/96 to 02/08) 98.94 % of the required inspection from the Mandatory Inspection List has been made.

3.6.4.2.5. CNA I Lessons Learned from National and International Operating Experiences

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

3.6.4.2 6. Other improvement activities in CNA I

- *Evaluation of the integrity of the Reactor Pressure Vessel (Details in chapter 3.14)*
- *Probabilistic Safety Analysis (PSA) (Details in chapter 3.14)*
- *Advances in the Severe Accident Management Program (Details in chapter 3.14)*
- *Improvements in the heat decay removal system (Details in chapter 3.19)*

3.6.4.2.7. CNE Design improvements derived from PSA Findings

- *A redundancy system was added to the dousing tank level measure in order to initiate low pressure stage of emergency water injection (Emergency Core Cooling system, ECCS), to improve the management of LOCA.*
- *Automatic closure of water injection valves with signal of low level in light water tanks of high pressure emergency water injection (ECCS) was implemented, in order to avoid air entrance in primary circuit when the entire inventory is consumed, during LOCA management.*
- *To improve the reliability of emergency power supply diesel generators, a redundancy system was implemented in the relay contacts involved in start up signal transmission to diesel generators.*

3.6.4.2.8. CNE Improvements in Plant Procedures derived from PSA Findings

- *Several Emergency Operating Procedures (EOP) were linked taking into account possible multiple failures.*
- *The EOP 5, referred to loss of service water, was modified in order to avoid diesel generator failure due to inadequate cooling. This procedure was also modified in order to implement emergency water supply to boilers in case of loss of feed water.*

3.6.4.2.9. Other Improvement Activities in CNE

For a better and orderly presentation, the following other CNE improvements activities are detailed in chapter 3.14 of this report

- *Refurbishment Program,*
- *Improvement in pressure tube inspection,*
- *Feeders inspections,*
- *Probabilistic Safety Analysis,*
- *Advances in the Severe Accident Management Program.*

3.6.5. 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. NASA 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.

3.6.6. COMPLIANCE WITH THE OBLIGATIONS IMPOSED BY THE CONVENTION

From the preceding considerations, it becomes evident that since the beginning of nuclear activities in the country, 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 Chapter 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 CNEA was created in 1950 by Decree No 10936, and one of its specific functions was to control the official and private atomic research carried out in Argentina.

Later on, different legal regulations determined CNEA's competence as Regulatory Body in the field of radiological and nuclear safety, particularly in those aspects concerning the individual and environmental protection against the harmful effects of ionising radiation, the safety of nuclear installations and the control of the use of nuclear material. The main legal regulations concerning these aspects are by Decree-Law No 22498/56, ratified by Act No 14467/58, and Decree No 842/58.

The mentioned Decree-Law also established CNEA's competence to issue the necessary regulatory standards and requirements for the permanent supervision of the activities related to radioactive materials, and to provide the necessary means to control the existence, commercialisation and use of materials related to peaceful applications of atomic energy.

On the other hand, Decree No 842 approves and puts into force the Regulations for Using Radioisotopes and Ionising Radiation, with the purpose of regulating the use and application of radioactive materials and their radiation emissions, their reactions and nuclear transmutations, as well as to sanction in cases of violation. The use of X-ray generators was excluded from CNEA's competence and entrusted exclusively to the Ministry of Health.

As a consequence of the increasing development of nuclear activity in the country, the functional independence of the Regulatory Body from CNEA's other activities became essential.

In 1994 the Government decided that the nuclear power production would be transferred to the private sector, preserving the regulatory function and control of nuclear activities at state level, and formally independent from promoters and users.

Based on these considerations, the National Executive Power, supported by Act No 23696/89 and by Section 99 Paragraph 1 of the Constitution, created the National Board of Nuclear Regulation by Decree No 1540/94, with territorial jurisdiction along the whole national territory. From this decree on, the National Board of Nuclear Regulation became the nuclear regulatory authority, performing all the regulatory and control functions of the nuclear activity, which formerly had been within the competence of CNEA's regulatory branch.

Act No 24804/97 (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.

Furthermore, Decree 1390/98 established 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 25,018/98 sets provisions that involve the ARN in the management of Radioactive Wastes and state that the ARN shall:

- *Approve the acceptance criteria and the transference conditions of the radioactive waste formulated by CNEA (application authority).*
- *Approve radioactive waste transference procedures, in particular 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.

This Act also affects indirectly the ARN, regarding the provisions of the resources necessary to exert its regulatory functions in the new facilities, due to the following Decrees:

- *Decree of the National Executive Power 981/05 (by which it instructs NASA, as Licensee, to conform the Unit of Management Atucha II, whose objective will be to carry out the Acts which are required for putting into operation CNA II),*
- *Decree 1085/06 (that maintains operational the regime restored for the execution of works on CNA II, granted to the Unit of Management Atucha II), and*
- *Decree 1107/06 (that declares of national interest the construction and operation of the CAREM for the nucleus-electrical generation of energy).*

3.7.2. NORMATIVE FRAMEWORK

3.7.2.1. INTRODUCTION

Act No 24804/97 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 some 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 stands for Regulatory Authority), has at present 61 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 standards are shown in Table 3.7.1.

Additionally, there is a permanent Regulatory Body activity, which is the standards review and the standards updating, with the aim of maintaining updated the normative system.

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 Authorisations
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

AR Code	Name
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

In addition, up to April 2007, seven regulatory guides were issued; 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*
GR1	*Dosimetric factors for external irradiation and internal contamination and emergency levels in food*
GR2	*Schedules of requirements for the transport of specified types of radioactive material consignments*
GR3	*Specific functional conditions to be verified by the specialized physician according to psychophysics performance score*
GR10	*Training programme of personnel for Type I installation*

In order to carry out the process of licensing of CNA II, the regulatory activities related with CNE's life extension, as well as to the future licensing of CAREM and the fourth NPP, the ARN has considered it opportune to intensify the revision of its regulatory standards and guides related to NPPs, as well as the elaboration of the necessary new standards.

The Regulatory Standard AR 3-1-3 was one of the main reviewed standards. This standard contains the Argentine acceptance criterion - from the point of view of safety - applicable to the design of NPPs. The recommendations produced by the experts designated to this task in order to improve its use, is under the consideration of the ARN Board of Directors.

Also in revision process is the standard AR-3.10.1 "Earthquake Protection", taking into account the "state-of-the-art" in the matter and the analysis of the following new regulatory standards which are in progress:

- *AR-3.10.2 "General Safety Criteria for Siting"*
- *AR-3.10.3 "Safety Criteria for Evaluation of External Events"*
- *AR-3.10.4 "Criteria for Determining the Potential Effects of the NPP on the Region"*

3.7.2.2. HARMONIZATION WITH IAEA STANDARDS

During 2006, ARN decided to initiate an harmonization process of their standards against the IAEA safety documents.

The detailed comparative analysis is well advanced, and comprise the criteria established in the ARN standards and the corresponding indicated in the IAEA Safety Requirements.

The comparative analysis is also extended to the recommendation that are contained in the Safety Guides, that would be important to be incorporated in ARN standards.

Until now it is possible to conclude that Argentinean standard are completely consistent with the IAEA corresponding documents, taking into account that ARN has adopted mainly performance criteria.

Nevertheless that was mentioned before, some improvements opportunities have been identified in no essential aspects of the ARN standards, which will be incorporated following the corresponding procedure.

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: deterministic and probabilistic ones. 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 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 is leader in using both approaches at international level, having adopted, 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, that is to say, of performance of systems, equipment and components. 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.

It is in such a context that the Licensee must convince the Regulatory Body that the installation is safe. The role of the latter is to be sceptic and critical, without proposing "how".

The before mentioned conviction implies an interaction of intense, continuous and personal nature, among professionals of the Regulatory Body and the Licensee, 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 could 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 prescription 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 ago and are coherent with the ICRP and IAEA recommendations.

Furthermore the ARN has contributed to formulate recommendations issued by international bodies (such as 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.

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 standards and requirements issued by the Regulatory Body. The non-compliance with any of the standards, conditions or requirements should be 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 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, which establish the conditions that the Licensee must fulfil at each stage.

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

The applicable 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 shall be presented together with the application for the Construction License (Standard AR 3.7.1). In particular the NPPs design must comply with the radiological criteria related to accidents (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 emphasised that the Licensee's capacity to carry out its responsibilities is evaluated 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. (Standards AR 3.7.1, AR 3.8.1 and AR 3.8.2).

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

The Operating License is a document by which the ARN authorises the commercial operation of a nuclear installation under stipulated conditions, which shall be fulfilled by the Licensee (Standard AR 3.9.1). The non-fulfilment of any of the imposed requirements without the corresponding ARN authorisation 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 authorises the ending of the NPP commercial operation and issues a Decommissioning License. In this document, conditions for the NPPs safe dismantling are established, being the Licensee in charge of planning and providing the necessary means for its fulfilment (Standard AR 3.17.1).

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

3.7.3.2.2. Periodic Safety Review and License Renewals

Until 2003, 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 Operating Licenses include similar requirements.

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 NPPs 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 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 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 this mayor 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

AR 0.11.1, AR 0.11.2, and AR 0.11.3 standards set the criteria and procedures to provide Individual Licenses and Specific Authorisations to the personnel who apply for licensable functions in nuclear installations. Besides, these standards establish terms and conditions according to which the ARN may issue such Individual Licenses and Specific Authorisations. *In addition, AR 0.11.3 standard establishes criteria on retraining of personnel for this type of installations.*

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 such 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 Authorisation is needed for its personnel, the Licensee submits the necessary documentation to the ARN. The "Consejo Asesor para el Licenciamiento del Personal de Instalaciones Relevantes" (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 Authorisation or for the renewal of the latter must fulfil a number of requisites concerning qualification, working experience, training, re-training 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.

Specialised 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.*

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

On the other hand, continuing the updating process of the normative system, the new Standard AR 0.11.3 "Training of the Licensed Plant Personnel" has been introduced. This standard establishes the specific requirements that have to be met for the plant staff training and the mechanisms for the evaluation of the training process.

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 standards, Licenses and requirements in force. All these activities are performed according to written procedures.

Act No 24804, authorises 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 detect eventual weak aspects, or identify possible ways of reducing personnel doses.*

- *Regulatory Audits are programmed in order to analyze organisation, operation and process aspects related to radiological and nuclear safety.*

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 evaluation.*

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

Act No 24804 enables the ARN to apply sanctions and to suspend or cancel the validity of Construction, Commissioning, Operating and Decommissioning Licenses, in the case of non fulfillment with standards, Licenses or any other regulatory requisites.

The sanctions to be applied in proportion to the seriousness of the fault and in relation with the potential or real harm, are in increasingly ordered: “warning”, “penalty”, “suspension” of the License or Specific Authorisation or even its “cancellation”.

It is worth mentioning that the ARN appeals to the consensus and conviction when regulatory demands are posed to the Licensee. It is the philosophy of the ARN to consider that a “sanction” is not a routine regulatory action but the ultimate measure to be adopted in a conflictive situation.

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. Such 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 situation and safety of the involved installations.

Act No 24804, sets that the ARN is in charge of the regulation and surveillance of nuclear activity 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 with State support.

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

Act No 24804 also gives the necessary legal competence to the 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 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 modified.

In order to guarantee a proper control level, such legal competence is complemented with a suitable technical capability. Namely, the ARN has the capacity to independently evaluate the construction, commissioning, operation and decommissioning of NPPs.

For this reason, since the beginning of the regulatory activities in the country, it was considered imperative to have qualified personnel with adequate knowledge and experience in order to preserve the ARN’s own independent criterion regarding every aspect of radiological and nuclear safety.

Besides, the ARN is empowered to contract specialists who can advice 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 authorisations.
- Independent execution of analyses and studies for the licensing process of nuclear installations.
- Development of technical and scientific aspects associated to radiological and nuclear safety.
- Training of personnel involved in radiological and nuclear safety, either belonging to the ARN or those working in installations, which perform practices under regulatory control.

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

Act 25,018/98 sets provisions that involve the ARN in the management of Radioactive Wastes. It states that ARN shall:

- *Approve the acceptance criteria and the transference conditions of the radioactive waste formulated by CNEA (as application authority).*
- *Approve radioactive waste transference procedures, in particular 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.

3.8.2. ARN ORGANISATIONAL STRUCTURE AND HUMAN RESOURCES

According to the provisions in Act No 24804, the ARN is managed and administrated by a Board of Directors, all members with adequate technical and professional background on the subject.

The ARN acts as an autarchic organisation *and reports directly to the Secretary of the National Executive Power. The Board of Directors consists of three members, a Chairman, and 2 Vice-Chairmen.*

The ARN organisation 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). The projects have a limited duration and once completed should be integrated into one or more activities, if the case requires it. A schematic chart of the ARN structure is shown in Figure 3.8.1.

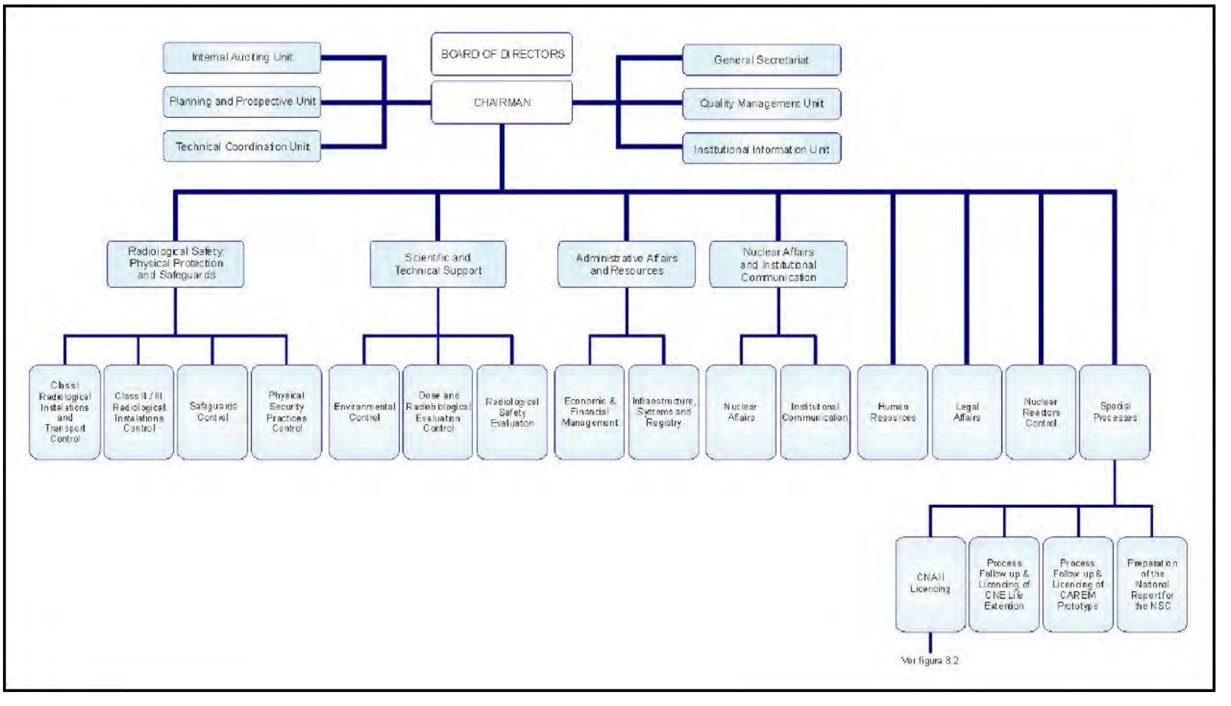


Figure 3.8.1 – ARN Organisation Chart

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

The Radiological Safety, Nuclear Security and Safeguards Division carries out regulatory inspections and assessments concerning Radiological Safety of Radioactive Installations (medical, research and industrial installations), Transport, Safeguards Control, and Nuclear Security Controls.

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

The Nuclear Reactor Control, Institutional Affairs and Non Proliferation, Legal Affairs, and Human Resources Departments report directly to the Board of Directors. The Nuclear Reactor Control Department is in charge of inspections and technical evaluations of the operating NPPs and research reactors.

The Special Processes Unit has recently been created to face the regulatory activities related to CNA II licensing, the CAREM licensing, CNE life extension and the feasibility study for a fourth NPP.

The Legal Affairs Section, the Internal Auditing Unit and the Planning and Prospective Unit advise the Board of Directors on legal aspects of regulatory management, on the use of economical resources of each project and on the ARN activity planning, prospective and management control.

The Institutional Affairs and Non Proliferation 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. This branch also controls the compliance of physical protection regulations applicable to nuclear materials and installations. Furthermore, it co-ordinates the institutional relations in the national and international sphere.

During the period covered by this report, the ARN has increased substantially its personnel, from 202 persons at the end of 2002 to 312 at the beginning of April 2007.

The important increase in personnel, mainly young professionals, is due mainly to the need to cover positions which have to be filled as the generation of specialised professionals reach retirement age, and also to cover the new tasks to be undertaken by the Special Processes Unit.

It is worth mentioning that of the 312 people working at the ARN, 80% are professional and technicians that undertake specialised technical tasks in areas of their competence or are under training programs, and 20 % perform administrative activities. Furthermore is the fact that 7% of the total personnel of the ARN hold high level positions or functions, and have a specialised training of about 20 years or more working in regulatory activities.

The whole staff is geographically distributed as follows: 69 % of the personnel at the headquarters in Buenos Aires City, 23% in the Ezeiza Atomic Centre and 5 % at the CNA II site. The remaining 3 % is constituted by 1% resident inspectors at the NPPs and 2% at the Argentine-Brazilian Agency for Accounting and Control of Nuclear Material (ABACC) with headquarters in Rio de Janeiro, Brazil.

3.8.2.1. RESOURCES ASSIGNED TO THE NUCLEAR POWER PLANTS REGULATORY CONTROL

3.8.2.1.1. General aspects

ARN is presently going through a period of a marked reduction of specialised personnel due to the fact that some are reaching retirement age and to the increasing offer of better-paid job opportunities from other industries.

The young professionals that do enter to ARN must undergo intensive training to attain the basic knowledge in the nuclear field. Nowadays, monetary considerations have a predominant importance over professional development in defying tasks and consequently it makes it much harder finding motivated young professionals with the desire to learn and progress in the scientific/technological sphere. In this context, the ARN is organising supportive courses for the incoming personnel and special courses for management directives.

Furthermore, the ARN is using knowledge management as a useful tool to transfer and keep up the knowledge standard among its professional personnel (see Chapter 3.8.2.5).Any deficiencies on the ARN response capacity due to lack of senior personnel is covered through work agreements and contracts with specialist, and domestic and foreign organisations.

ARN not only trains its own personnel on radiological protection and safety of radioactive and nuclear sources, it also trains technical people from other national and international institutions.

For what was mentioned before, ARN has the necessary resources and infrastructure to carry out its mission and to reach its objectives.

3.8.2.1.2. Human Resources assigned to operation control

About 55 persons are involved in regulatory activities related to NPP control. The percentage distribution of human resources assigned directly to NPPs inspections and safety assessments *between* 1998 and 2006 was as follows:

Inspections and evaluations 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 who study particular issues regarding the installation safety.

This distribution of efforts has experienced little variation and the fraction of efforts invested in inspections and evaluations were relatively constant during the mentioned period. As of that date and as has already been mentioned, the increase in the licensing tasks for CNA II, CNE life extension and CAREM licensing as well as the pre-feasibility studies for the Forth NPP, have implied a temporary redistribution of the efforts dedicated to the operating NPPs, reducing somewhat indirect support activities but maintaining the man hours directly applied to the inspection of the installations.

3.8.2.1.3. Human Resources assigned to construction and commissioning control

The human resources used by the ARN during CNA I construction and commissioning stages have been different to those assigned to CNE and to those that are assigned to CNA II for the same stages. This was due to the different circumstances in which those activities were developed and to the different ARN experience in such occasions.

In the case of CNA I the role of Independent Authorised Inspector, prescribed by the ASME code, was performed by two entities: Technischer Üeberwachungs 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. Such entities carried out the verifications of:

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

In the case of CNE, in order to co-ordinate the tasks related to licensing, during construction and commissioning ARN organised a special committee, called Executive Committee for CNE Licensing. The main functions of this committee, were:

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

This committee performed the safety evaluations during the plant construction and commissioning on its own or by contract with third partners. The seismic re-evaluation of the installation was significant among them.

In the case of CNA II, ARN is commissioning a second-generation NPP whose Construction Licence was granted in July 1981 and whose construction has stretched over nearly three decades and has not finished yet. ARN has analysed on various occasions the commissioning aspects of CNA II, and concluded that it is feasible as long as the pertinent regulations are complied with (see more details in Chapter 3.18). ARN has formed an ad hoc organisation for the commissioning of CNA II whose structure is shown in Figure 3.8.2.

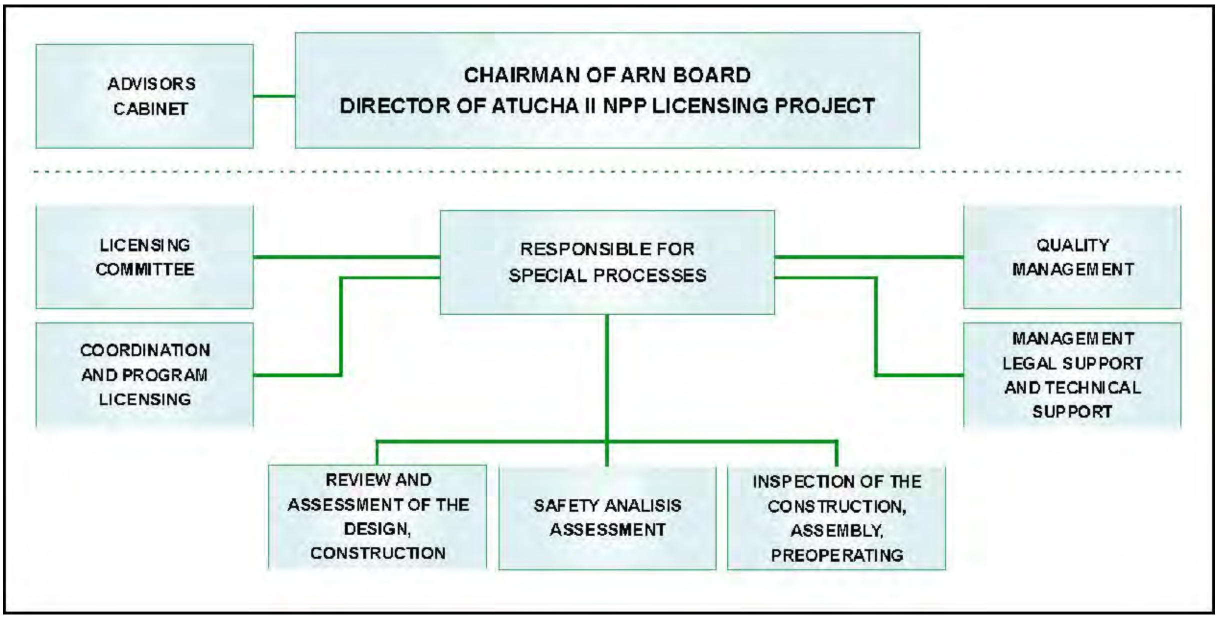


Figure 3.8.2 – Structure of the organisation in charge of the commissioning of CNA II

Since April 2007 the composition of this ad-hoc organisation consists of 22 Professionals, 4 Technicians and 3 Administratives. Fifty five percent (55%) of these people work on site, while 45% work at headquarters. The quantity, composition and qualification of the personnel of this organisation, may change in the future according to the requirements of the commissioning activities.

Furthermore, four professionals have been appointed for the regulatory activities corresponding to the life extension of CNE, while for the commissioning of CAREM and for the preoperational studies of the fourth NPP, two professionals have been allotted. These professionals receive technical and logistic support from other areas of ARN.

3.8.2.2. ARN PERSONNEL QUALIFICATION

As part of their training, ARN professional personnel attend Post Graduate Courses on Radiological Protection and Nuclear Safety. Subsequently, those professionals assigned to NPPs inspections receive a wide and complete specific training in NPPs. On the other hand, inspectors also participate in periodic courses and seminars organised by NPPs for their own personnel, as well as in courses and seminars organised by different domestic and international organisations.

Furthermore, ARN´s specialised personnel participates in activities of organisations such as ICRP, UNSCEAR, as well as OSART and INSARR missions of the IAEA in different foreign NPPs and research reactors.

3.8.2.2.1. Maintaining competence of the Regulatory Body

In the frame of the National Plan for the Modernization of the Public Sector adopted by the National Executive Power, in January 2003, the ARN signed a Program Agreement with the following strategic goals:

- To maintain and improve the regulatory quality and efficiency; in particular to address the challenge of the gradual loss of specialized human resources with the scientific and technical knowledge required to guarantee the quality of the regulatory decisions and of the control activities.
- To strengthen the institutional viewpoint.

Under this agreement, the ARN undertakes to improve its work and the quality of its nuclear regulatory function through the strengthening of its management tools and the willingness to introduce required changes to that end.

The National Executive Power undertakes to support ARN's activities to facilitate the introduction of the required changes to achieve the above-mentioned goals. This support includes – inter allia – the approval of administrative decisions oriented to obtain financial aid to improve the quality, integration and management of the regulatory function and the technical assistance to strengthen the ARN institutional image.

Furthermore, this agreement comprises the establishment of quantifiable and verifiable results and a flexible system to achieve an improvement in the areas of efficiency, efficacy and quality management (Law 25.512, Regulation of the Public Administration Sector).

At the end of 2006, the activities of the Program Agreement were at different stages of implementation:

1. *Certification of Laboratory under ISO/IEC 17025:2005 Standards: In February 2007 the Environmental Control Laboratory was certified.*
2. *Certification of training course: In 2003 the post Graduate Course on Radiological Protection and Nuclear Safety, which until then was given as one unit, was divided into two: a Course on Radiological Protection and Safety of Radioactive Sources and a Course on Nuclear Safety. In March 2007, both courses were certified under ISO 9001:2000 Standards.*
3. *Argentina was designated Regional Centre for Training on Radiological Protection and Nuclear Safety for Latin America and the Caribbean (announced at IAEA´s Board of Governors meeting in December 2006 as a formalisation of a procedure in process). On March 2007 the regional courses were certified.*

At an advanced implementation stage are:

- *The new organizational structure,*
- *The Quality Management System,*
- *The Press and Communication office,*
- *The incorporation of young professionals,*
- *Certification of the procedure "Protection against Ionizing Radiation in the Transportation of Nuclear and Radioactive Material". In August 2007 the certification pre-auditing will take place and in October the Certification procedure under the ISO 9001:2000 Standard will do so too.*

At a less degree of development are:

- *The retrieval, transference and refreshing/updating regulatory knowledge (Knowledge Management),*
- *The plan for an institutional career,*
- *The strategic communication plan,*
- *The integrated trainig plan.*

3.8.2.3. QUALITY MANAGEMENT SYSTEM IN THE REGULATORY BODY

The ARN, as a "de facto" and "de jure" independent authority, has always taken actions with the objective of improving the quality of its regulatory performance. The following achievements have been accomplished:

- An annual work-plan and its management control, as well as the permanent development of technical and administrative procedures applicable to regulatory activities,
- Continual improvements of the existent normative system by means of writing new standards and the review of the existing ones,
- Annual training programmes for technicians and professionals,
- ARN Annual Reports issued for the general public, sent to the National Legislative Body (Honorable Congreso de la Nación),
- Publication of technical and scientific reports.

In 2002, the ARN Board of Directors decided to strength the quality management activities by planning its development and implementation. The support of external advisors involving the National Technological University of Buenos Aires (UTN-BA) was procured. An agreement between ARN and UTN-BA was signed. The agreement goal was to achieve a well-structured quality management system, focused on external and internal stakeholders and based on the "Continual Improvement Approach". The UTN-BA agreement was fulfilled and finished in 2005.

3.8.2.3.1. Quality Management System

3.8.2.3.1.1. Implementation

For an appropriate implementation of the quality management system, ARN created in 2005 the Quality Management Unit, which depends on the Board of Directors (see Figure 3.8.1).

ARN has established, documented and implemented the Quality Management System according to the requirements established in the ISO 9001:2000 Standard. The actions and requirements are described in the Quality Manual (MC-ARN). In this document the ARN Board of Directors declares and communicates the Quality Policy and Commitment.

Additionally the Board of Directors decided to initiate the discussion, interpretation and implementation of the IAEA Safety Requirements GS-R-3-Management System for Facilities and Activities.

The Quality Management System is implemented by applying a processes approach. Seven regulatory processes and four support processes have been identified and they are represented in the following chart:

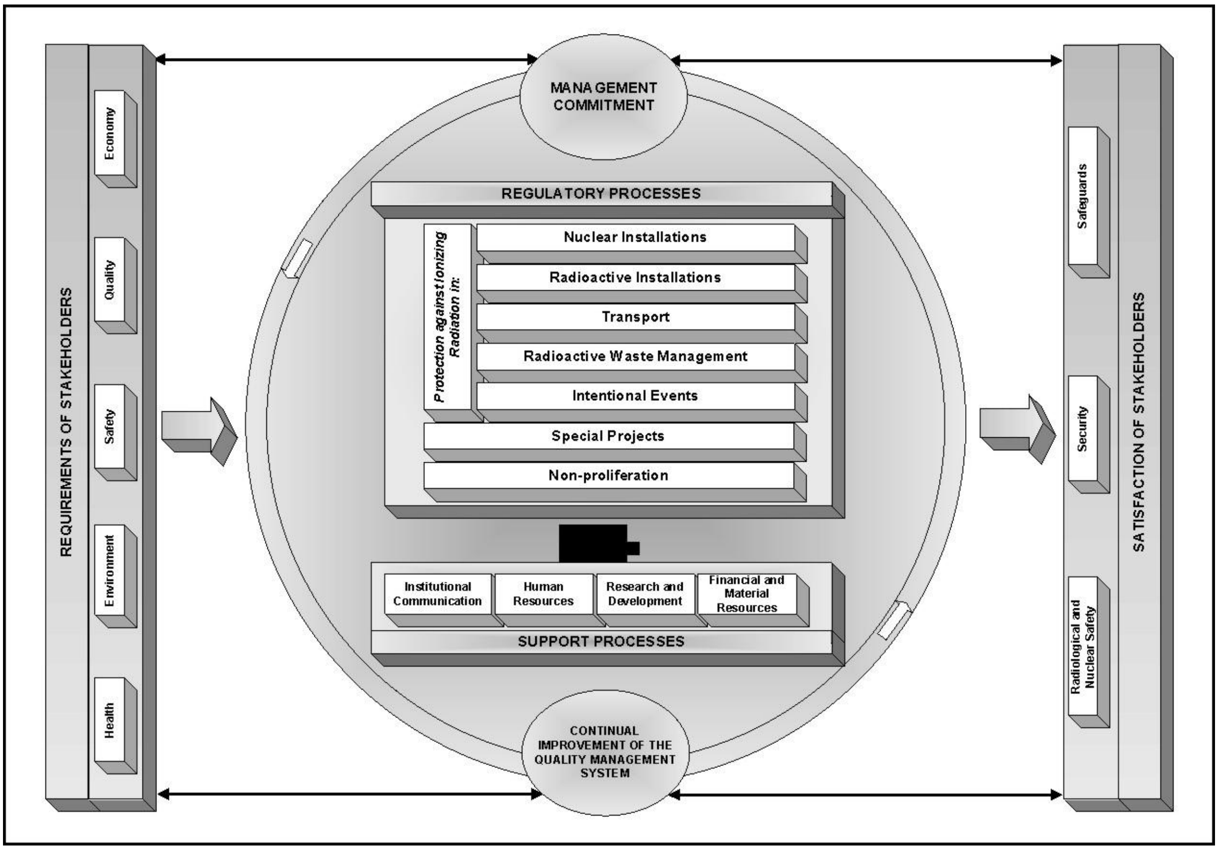


Figure 3.8.1

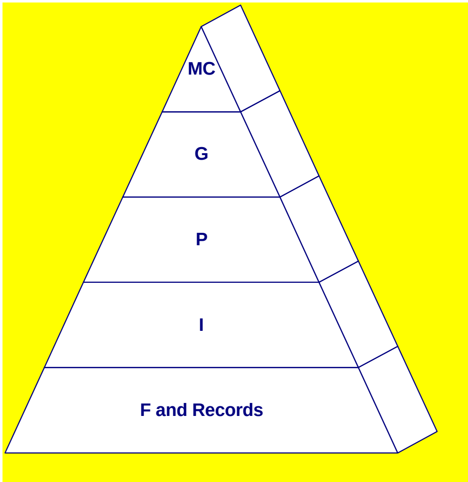
The sequence and interaction of these processes are established and represented in charts (layout).

Each process is described in a Process Letter where the objectives, inputs, outputs, checkpoints, associated documents, performance indicators, non-conformances and corrective actions are considered.

The analysis and measurements of these processes are carried out by self-assessments. Effectiveness and Efficiency Indicators are been defined.

3.8.2.3.1.2. Documentation Requirements

The Quality Management System is based on a solid documental structure, which involves:



Quality Manual (MC-ARN)

General Procedures (G)

Specific Procedures (P)

Working Instructions (I)

Forms (F) and Records

Up to June 2007 there were ninety-one (91) approved documents and twenty-seven (27) documents are being prepared at present. Documents and Records control and technological Information security controls have been implemented.

3.8.2.3.1.3. Satisfaction of stakeholders

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

3.8.2.3.1.4. Quality Management System Planning

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

The Quality Management System changes are performed in a planned way, so as to assure its continuity, effectiveness and efficiency.

3.8.2.3.1.5. Measurement, Analysis and Improvement

ARN carries out quality self-assessment in order to:

- *demonstrate the products and processes compliance with stakeholders requirements,*
- *ensure the Quality Management System compliance,*
- *continuously improve the effectiveness and efficiency of the Quality Management System,*
- *identify opportunities for improvement,*
- *verify that corrective actions are carried out.*

The self-assessments are performed by qualified personnel. They are independent of the area which is assessed. During 2006 and until June 2007, eleven (11) assessments were carried out; 17 non-conformances, 10 observations and 63 opportunities for improvement were identified.

3.8.2.4. COMMUNICATIONS WITH THE STAKEHOLDERS

ARN has implemented communication mechanisms with the stakeholders concerning:

- *legislation, standards, regulatory guides, licenses, permits, authorisations, and*
- *feedback regarding consultations, complaints, opinions and opportunities for improvement.*

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

In ARN's Website (www.arn.gov.ar), regulatory information, regulatory standards, laws and acts, permit and operating licence information, annual reports and general public communications among others, can be found.

3.8.2.5. KNOWLEDGE MANAGEMENT

In early 2006, the ARN started an initiative on regulatory knowledge management to give consideration to the generation gap, the loss of knowledge resulting from retiring experts, and the need to transfer such knowledge to younger generations through training. As a result two projects were initiated so far.

First, to find and turn both tacit and implicit knowledge, explicit. For this reason an in-depth interviewing approach was used with ARN's experts. Eighteen experts have been interviewed so far and most of the tape-recorded interviews have been transcribed. The results of these interviews have been used to find possible projects between the experts and the new workers and to strengthen the training programs in the organization. The methodology used was the "History of the Learning Process".

Second, with the same objective, in 2007, a mapping process was initiated on one regulatory knowledge domain of the organization.

Finally, the ARN participated in the last International Conference on Knowledge Management in Vienna in June 2007, presenting ARN's knowledge management initiative.

3.8.3. FINANCIAL RESOURCES

The effective fulfilment of the regulatory objectives, requires the ARN to have an efficient structure and adequate personnel together with the necessary economical resources. Concerning this matter Act No 24,804, establishes that such resources shall 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 installed NPP, which must be annually paid 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 fees what is set forth in the “Regime for the Regulatory Rate for New NPPs” is applied. The fees cover the cost of regulatory activities during the NPPs construction, assembly, preliminary tests and commissioning stages.

The 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 2004 and 2007 are shown in Table 3.8.3. An important increase in the 2007 budget can be observed that allows the ARN to accomplish the licensing of the new projects mentioned before and continue with its normal tasks. The total budget during 2007 is composed 58.1 % from National Treasury, 37.7 % from annual regulatory fees and goods or resources assigned according to applicable laws and regulations, and 4.2 % from donations.

Table 3.8.3– ARN comparison budget for financial years 2004 and 2007

Item	$ (in thousands of Argentinean pesos)	
	2004	***2007***
1- Personnel	10,415	*24,388*
2- Support goods	392	*1,540*
3- Services	3.552	*12,019*
4- Equipment	1,307	*2,042*
5.1- Fellowships	262	*350*
5.9- Transfers	5,.556	*6,293*
9-Other expenses	305	*450*
TOTAL	21,789	*47,082*

The ARN 2007 percentage budget distribution is shown in figure 3.8.4.

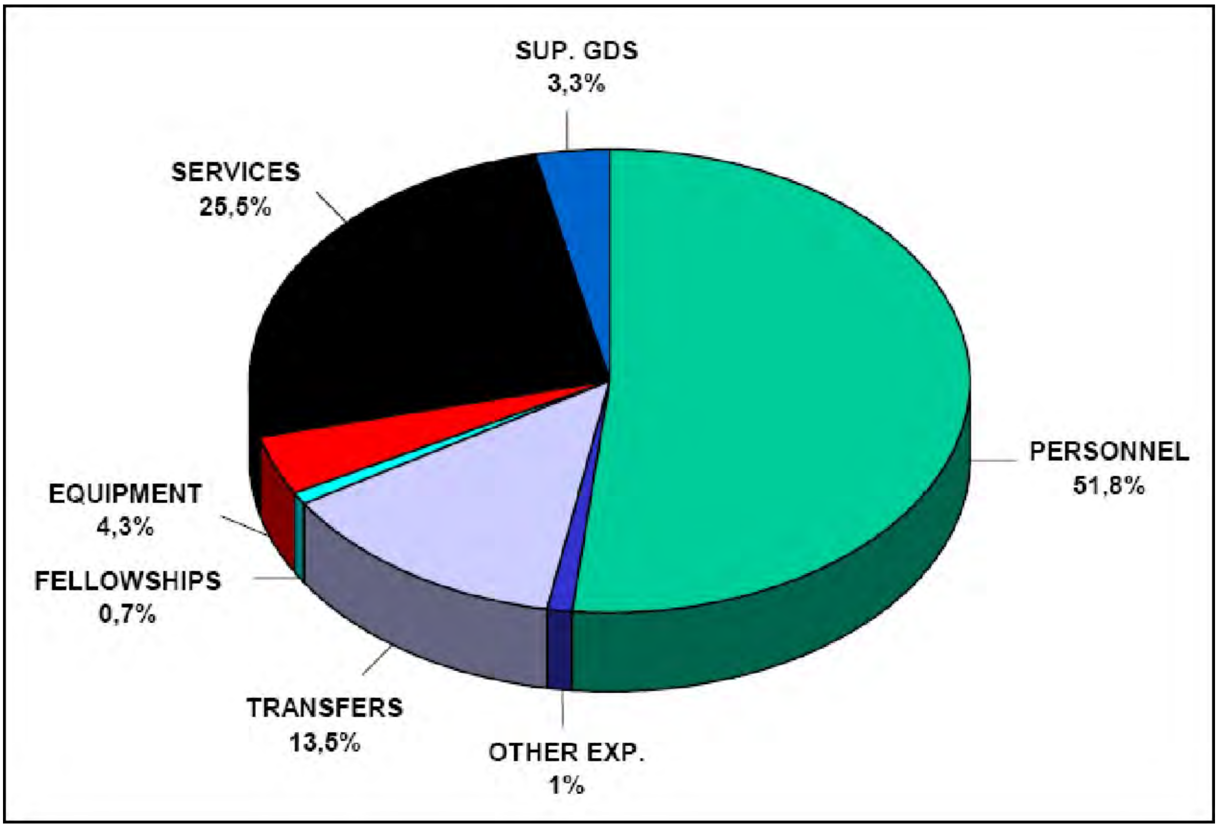


Figure 3.8.4.- ARN 2007 percentage budget distribution:

Additionally, the financial resources distribution for 2007 by tasks and by type of inspection are shown in Figures 3.8.5 and 3.8.6.

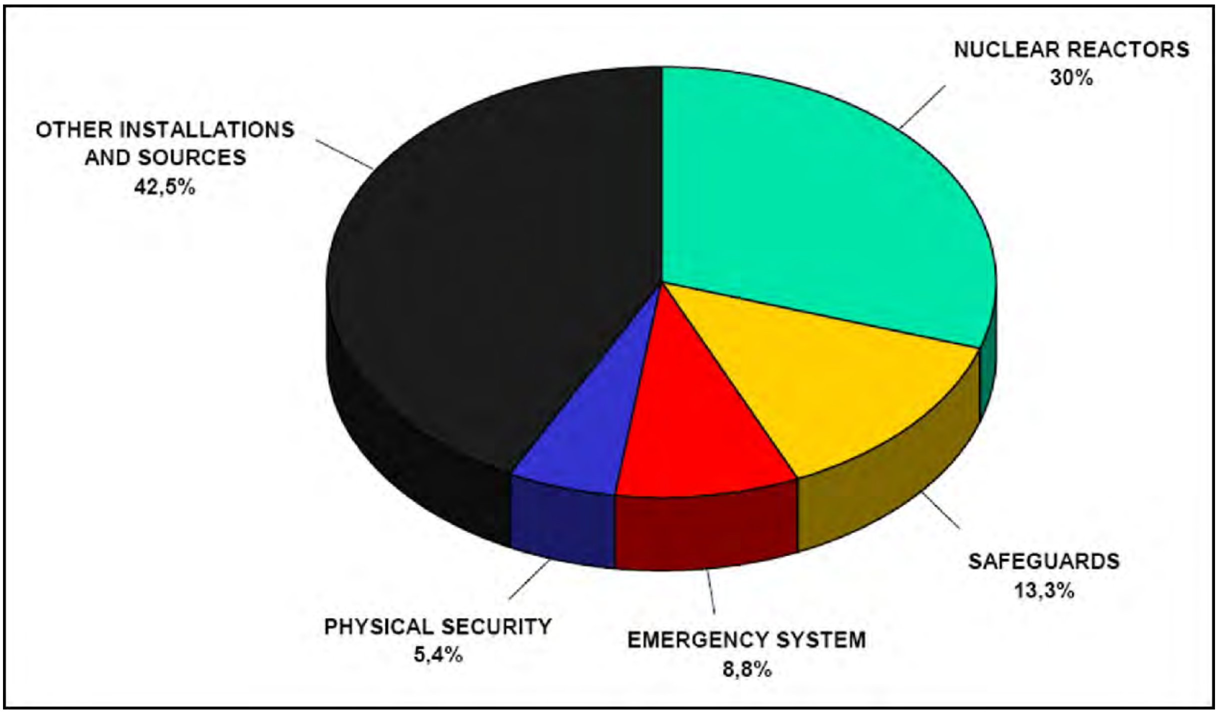


Figure 3.8.5.- Budget distribution in 2007 work plan by tasks

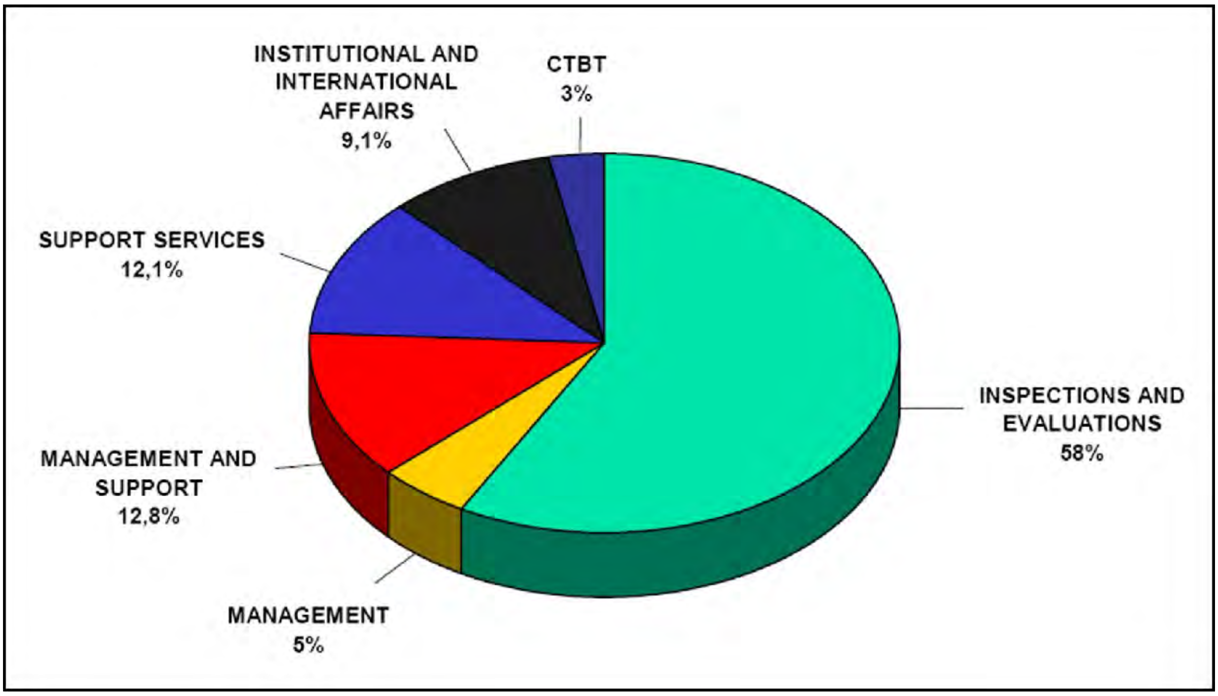


Figure 3.8.6.- Budget distribution in 2007 work plan by type of inspection

3.8.4. RELATIONSHIP WITH OTHER ORGANISATIONS

It is worth mentioning that while performing its regulatory functions, the 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 Convention, the relationship between the ARN and other organizations remains the same as far as regulatory activities are concerned. It has continued to participate in the Forum of Ibero-American Nuclear Regulators and the Network of Regulators of Countries with Small Nuclear Programs (NERS).

Its activity has been particularly intense in the Ibero-American Forum, and participated, since 2003 until now, in the following meetings:

EVENT	***PLACE***	***DATE***
Plenary	*Buenos Aires, Argentina*	*May, 2003*
Plenary	*Río de Janeiro, Brazil*	*Jan, 2005*
Technical Committee	*Vienna, Austria*	*Mar, 2005*
Plenary	*La Habana, Cuba*	*Jan, 2006*
Technical Committee	*Río de Janeiro, Brazil*	*Apr, 2006*
Plenary	*Madrid, Spain*	*Jun, 2006*
Plenary	*Cancún, Mexico*	*Jul, 2006*

Regarding NERS, after a pause of a few years in its activities, the ARN reinitiated in 2007 its participation.

The relationship between the ARN and domestic and foreign organisations is carried out through agreements which rule the co-operation provided by such institutions. A list of the new agreements detailing their respective purpose is shown in Table 3.8.7 for domestic organisations, and in Table 3.8.8 for foreign organisations.

Table 3.8.7 – Agreements with domestic organizations

AGREEMENT COUNTERPARTS	***AGREEMENT OBJECTIVES***
Agreement between the ARN and the Universidad Nacional de Tucumán. Feb. 2007	*Establish a mutual cooperation system for the promotion and execution of tasks related to education, investigation, culture and technique areas.*
Agreement between the ARN and the Fundación Escuela de Medicina Nuclear (FUESMEN). Nov. 2006	*Collaboration in areas of common interest related to applied radiological protection in medicine. Projects and studies in common.*
Agreement between the ARN and the Ejército Argentino. Nov. 2006	*Its principle objective is to establish training activities for human resources and other areas of mutual interest as defined by the parties.*
Framework Agreement between the ARN and the Ministerio de Salud de la Nación. August 2006	*Collaboration in regulatory aspects of the use of ionizing radiation.*
General Cooperation Agreement between the ARN and the Asociación Civil Ciencia Hoy (Proyecto RETINA).	*Collaboration in the development and use of communication technologies and its use in advanced networks.*
Agreement between ARN, CNEA, the Ministerio de Ambiente y Obras Públicas de la Provincia de Mendoza, the Municipalidad de la ciudad de San Rafael and the Universidad Nacional de Cuyo. December 2005	*Establish and undertake environmental monitoring programs for Uranium, Radium 226 and Radon 226 in the area of San Rafael.*
Framework Agreement between the ARN and the Universidad Tecnológica Nacional Facultad Regional General Pacheco. Jun 2005	*Maintain a mutual cooperation system for the promotion and execution of tasks related to education, investigation, culture and technique areas.*

Table 3.8.8 – Agreements with foreign organizations

AGREEMENT COUNTERPARTS	***AGREEMENT OBJECTIVES***
"Agreement between ARN and The National Nuclear Regulator of The Republic of South Africa for Technical Co-operation and Exchange of Information in The Regulation of Nuclear Safety" Feb. 2007	*To establish a framework for the Parties to co-operate in matters of mutual interest concerning regulatory aspects in the uses of nuclear energy.*
"Technical Services Agreement OP42086RR between ARN and Battelle Memorial Institute" Oct. 2006	*Provide ARN with technical services for the Atucha II NPP project.*
"Research Agreements No. 0ZF90; 0ZG22 and 0ZF91 between ARN and Purdue University". Nov. 2006	*Agreements signed within the CAMP framework, related to specific work areas for the commissioning of CNA II NPP. The focus of the agreements is the modification of computer programs that are owned and supplied by the US government.*
GRS – ARN "Arrangement for Consultancy Work and Services for Licensing of Atucha II" under the framework of the agreement Sept. 2006	*GRS will, on request of ARN, perform support for topics selected by ARN in the frame of the review of the licensing process, design, construction, assembly, preliminary tests and start-up for Atucha II nuclear power plant.*
"Implementing Agreement between the US-N RC and the ARN in the area of Cooperative Thermal-Hydraulic Code Applications and Maintenance". June 2006	*To allow ARN´s participation in this NRC´s program of cooperation in the field of reactors and plant system research (CAMP), helping to ensure internationally the safety of reactors and exchanging technical information in the area of nuclear regulatory research.*

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

Argentina is one of the few countries with members in the four technical committees and in the commission working within IAEA's Secretariat-established process for the preparation and review of safety standards. These are:

- *Radiation Safety Standards Committee (RASSC),*
- *Nuclear Safety Standards Committee (NUSSC),*
- *Waste Management Safety Standards Committee (WASSC),*
- *Transport Safety Standards Committee (TRANSSC),*
- *Commission on Safety Standards (CSS).*

An officer from Argentina is also present in the Standing Advisory Group on Safeguards Implementation (SAGSI).

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

3.8.5. 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 organism 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 licence and shall take the appropriate steps to ensure that each such licence 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. 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 authorisation, 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 authorisation was granted.

Though these concepts are still essentially valid for smaller 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 receive specialised training and have their own individual license. Besides, the training requisites for the whole operation personnel have been increased (see Chapter 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 organisation 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 organisation known as Licensee is Nucleoeléctrica Argentina S.A. (NASA), company in charge of the operation of the NPPs. The regulatory standards AR 0.0.1 and AR 10.1.1 establish its responsibilities, being some of the significant ones:

- The Licensee should 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 and procedures 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 shall 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 tasks totally or partially, but it maintains the whole responsibility.
- In each NPP the Licensee shall appoint a person of its own body, named Primary Responsible, who will be assigned the direct responsibility for the radiological and nuclear safety of such installation, 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 shall 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 authorisation.
- 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, shall 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 17048, 1966) has determined for the licensee. Act No 24804, 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 AR 10.1.1 standard 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. REGULATORY CONTROL ON THE FULFILMENT OF THE LICENCEE RESPONSIBILITIES

Since 1958, the Regulatory Body controls the fulfilment of standards, licenses and authorisations granted. In order to verify if the Licensees fulfil their corresponding 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 organisation chart must be reported to the Regulatory Body thirty days before the date of execution. Besides, these modifications are usually known by the Regulatory Body either through the routine meetings held with the Licensee or via the resident inspector's report.
- The AR 0.11.1 standard sets the requisites to be fulfilled by the NPP personnel in order to obtain the corresponding individual licenses and specific authorisations, according to Section 3.7.3.2.3.
- The procedure of issuing individual licenses and specific authorisations 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 authorisation 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 authorisation 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 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.
- Moreover, the Regulatory Body performs a permanent follow up of the Technical Revision Committee and the Internal Safety Advisory Committee minutes (see Chapter 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, particularly in its last paragraph.

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

As indicated in Chapter 3.7, Act No 24804,1977 enables the ARN to apply sanctions and to suspend or cancel the validity of construction, commissioning, operating and decommissioning licenses, in the case of non fulfillment of standards, licenses or any other regulatory requisites. In almost all cases, regulatory promotion and verification were adequate mechanisms to verify Licensee compliance with regulatory requirements.

3.9.4. 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 should 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 shall be carried out in accordance with a coherent regulatory system of principles, criteria and safety policies applied in the country for decades. Argentina, on occasions, contributed to international organisations 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 organisations are involved in the compliance with the above-mentioned principles:

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

Thanks to both principles, these two institutions coexist, being at the same time, completely independent one from the other.

The ARN establishes and applies a regulatory frame work to all nuclear activities developed in Argentina, with the following purposes:

- Protect people 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-authorised purposes according to Act No 24804, 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 unauthorised 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 Authorisations,
- Independent performance of studies and assessments about 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 subjects related to radiological and nuclear safety, safeguards and physical protection, for those members of the staff who are responsible for the safety of radiological practices subject to control and for those 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 NASA) 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 respect, and according to NPPs operation, NASA takes the following documents into account:

- Operating License,
  - Safety Report,
  - Policies and Principles Manual,
  - Operating Manual,
  - Maintenance Manual,
  - Quality Assurance 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 organisation'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.

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 advises to the Plant Manager, and its members are choiced for their knowledge and experience. This advise is regarding the actions to be followed in, for instance:

- Outages,
- Safety related incidents,
- Modifications to the installation (safety or safety related systems),
- Abnormal situations,
- Periodic evaluation of the installation performance,
- Periodic evaluation of the training personnel program,
- Emergency plan, 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 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 plant 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, in the case of the regulatory control principle and the Licensee, in the case of the responsibility 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 also included in the program for renewal of personnel Specific Authorizations.
- Review of the SC attitudes during inspections by ARN specialists.
- Review of trends in event reports, corrective action effectiveness and measures implemented to prevent safety problems.
- Review of trends for safety performance indicators.
- Assessment of minor event responses reported by the Licensee to detect organisational weaknesses and inadequacies.
- Increasing use of PSA for plant safety management.
- Improvement in the relationship between ARN and NASA.
- Efforts to improve the safety of the NPPs have the highest priority in both ARN and NASA.

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 that the Regulatory Body can promote Licensee's SC own proficiency, which includes, among other, attitudes: professionalism, teamwork, organisational 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 rentability of the mentioned plants.

As clear examples to this commitment is the fact that the NPPs comply with the defence in depth principle, or the case of operational situations in which a deviation from normal operating conditions occurs, and the decision to shutdown the NPPs is taken (rule that has been observed along the NPPs' lifetime).

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 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 the ARN and NASA 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 backfitting related activities in the installations are also analysed. The conclusions and recommendations are issued as minutes.

On the other hand, it should be pointed out that the installation personnel promote the dialogue and their active participation with the ARN personnel and viceversa.

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 or even improvement of working position (concerning both the 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 organisation and management (organisational factors) as the root and direct cause of events.*

The regulator/operator relationship has shown an important advance in safety culture. As for 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 of NASA and ARN specialists aiming at facilitating and improving the compliance with general and specific requirements, evaluating, in addition, the operational situation of the NPP (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 organisations of nuclear operators: the Candu Owners Group (COG) and the World Association of Nuclear Operators (WANO). Both organisations 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 performed by IAEA 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 organisations (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

As was mentioned in the 3rd National Report, during the period 2001-2003 there were macroeconomic difficulties which resulted in a recessive process, which produced a negative variation of the internal gross product in 2002. A strong devaluation of the Argentine peso took place (after maintaining a parity 1$= 1 US$ for more than one decade with an insignificant inflation rate). The currency devaluation, added to recession, had serious effects, among which we can mention those that affected the electrical market. The devaluation was followed by a low but sustained inflation, which affected the social economy since the wages remained without change.

Between 2004 –2006 the argentine economy had a significant growth, with an annual rise of 8% in GDP. Annually inflation rate was approximately 10%, while the currency exchange was 3$ = 1u$S.

During this period, the electric tariff for industries accompanied the rise in production costs, while the residential tariffs remained the same.

Consequently, the tariffs increase were below the increase of the generation costs. These costs rose due to the increase in the cost of national provisions and services and in a larger extent, of foreign supplies.

The difference between the real generation costs and the tariffs applied have been absorbed by the National Government using different methods for mitigating its effects.

The freezing of tariffs also reached the fossil fuel market that was also deregulated like the electrical market. It is important to keep in mind that solid propellants – particularly natural gas – provide up to 45 % of the electrical generation.

The increase of electrical demand in this period was of 7% annually, while the installed capacity had no changes. In order to diminish this deficit, the Government decided in 2006 the construction of two combined cycle plants through a joint investment -50% state-owned capital, 50% private capitals; it also announced Atucha II NPP completion, the elevation of the Yacyretá hydroelectric Power Plant and several other works like the extension of the High Tension Electrical Grid, and an increase in the transport capacity of the gas pipelines.

In spite of the lack of investment, the increase of the electrical demand was covered with the excess of reserve of the system, not allowing the storage of the electrical supply.

In Figure 3.11.1. we can find the electrical generation from 1975 to 2006, discriminated in hydroelectric, thermal and nuclear.

Considering the tariffs framework of minimum costs, nuclear generation was not affected. The production goals were met, all the programmed maintenance, improvements and changes in design for updating facilities, took place.

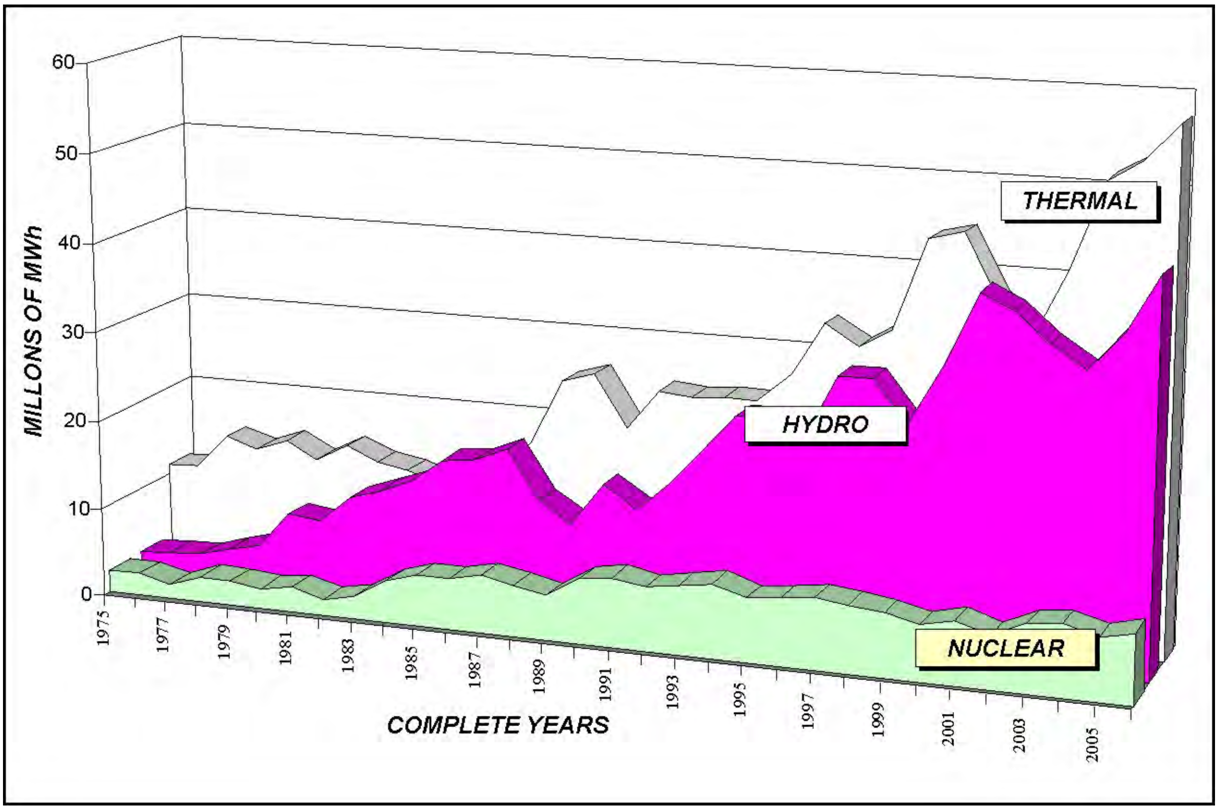
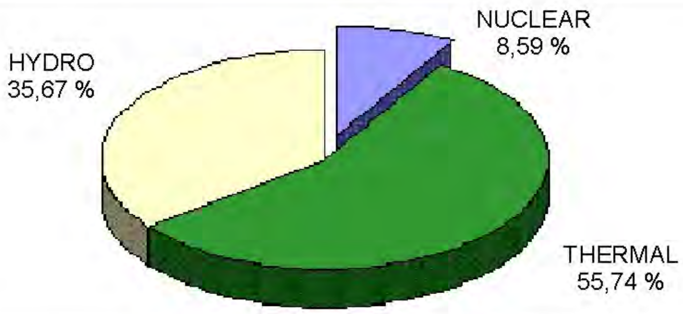


Figure 3.11.1 - Electrical generation in Argentina from 1975 to 2006

3.11.2. ELECTRICAL GENERATION AND ECONOMICS

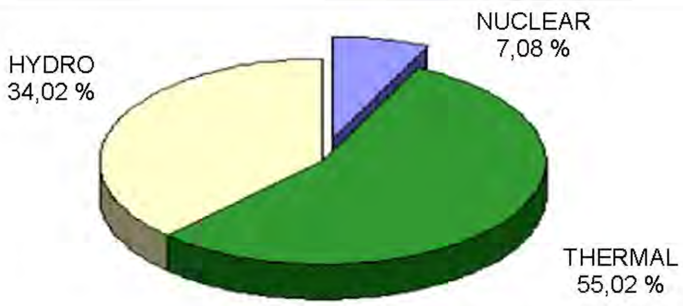
3.11.2.1. TOTAL ELECTRICAL GENERATION PERIOD 2004 - 2006

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



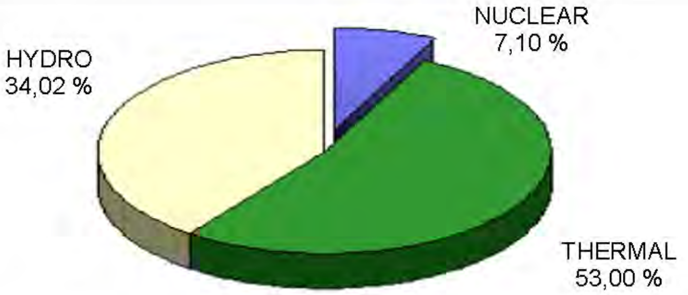
NUCLEAR	**7.868,6**	**GWhe**
THERMAL	*51.060,7*	*GWhe*
HYDRO	*32.674,0*	*GWhe*
TOTAL	*91.603,3*	*GWhe*

Figure 3.11.2 - Gross Energy Generated in 2004



NUCLEAR	**6.857,0**	**GWhe**
THERMAL	*53.280,5*	*GWhe*
HYDRO	*36.699,7*	*GWhe*
TOTAL	*96.837,2*	*GWhe*

Figure 3.11.3 - Gross Energy Generated in 2005



NUCLEAR	**7.690,9**	**GWhe**
THERMAL	57.400,8	GWhe
HYDRO	43.212,6	GWhe
TOTAL	108.304,3	GWhe

Figure 3.11.4 - Gross Energy Generated in 2006

3.11.2.2. NUCLEAR GENERATION 2004, 2005, 2006

Performance and energy fees are shown in tables 3.11.2, and 3.11.3.

Table 3.11.2 - Performance

	2004	**2005**	**2006**
Gross Energy (MWh)	*7.868.603*	*6.857.026*	*7.690.909*
Load Factor (%)	*89,13*	*77,89*	*87,36*
Installed Nuclear Power (%)	*4,36*	*4,36*	*4,18*
Generated Nuclear Power (%)	*8,59*	*7,08*	*7,10*

Table 3.11.3 - Energy Fees (annual average in $/MWh)

YEAR	*2004*	*2005*	*2006*
VALUE	*38,41*	*52,73*	*78,67*

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, NASA requested authorization to develop a 5 year program in order to incorporate new professionals. This Programme covers the years 2002-2007, and is currently under way.

Between 2004-2006 personnel at NASA increased by 138 employees. (See Table 3.11.1.).

Qualification programs and re-training of the operations personnel and technical, administrative and professional staffs of both NPPs took place, as is shown in the table 3.11.1.

Table 3.1.1 Personnel 2004, 2005 and 2006 by work area and specific knowledge

	Year	**CNA I**	**CNA II**	**CNE**	**MAIN BRANCH**	**TOTAL**
Professionals	*2004*	*74*	*38*	*102*	*90*	*304*
	2005	*78*	*38*	*103*	*98*	*316*
	2006	*79*	*57*	*102*	*106*	*344*
Technicians	*2004*	*275*	*81*	*379*	*55*	*791*
	2005	*289*	*80*	*383*	*61*	*812*
	2006	*295*	*120*	*381*	*65*	*862*
Administratives	*2004*	*62*	*32*	*85*	*34*	*213*
	2005	*65*	*32*	*86*	*37*	*219*
	2006	*66*	*47*	*85*	*40*	*239*
TOTAL	*2004*	*411*	*153*	*566*	*179*	*1309*
	2005	*431*	*150*	*571*	*196*	*1348*
	2006	*441*	*226*	*569*	*211*	*1447*

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

As the schedule for the completion of CNA II is already established, there is a short time to prepare the new personnel; this forced NASA to start the training as soon as possible.

Having all these requirements in mind, several training courses have been organized through specialized centers, like the Instituto Dan Beninson (Universidad National de San Martin) which provides training in nuclear reactors and fuel life cycle .

Several negotiations are currently taking place with CNEA, the Instituto Balseiro and other international companies, in order to obtain support for this training, either in preparing the materials, or in providing the simulation tools.

3.11.2.4. CURRENT EXPENSES. 2004, 2005, 2006

Table 3.11.4 and Figure 3.11.5 show the evolution of Operating and Maintenance costs

Table 3.11.4. Evolution of O&M costs (pesos $ x 1.000.000) period 2004 - 2006

TITLE / YEAR	***2004***	***2005***	***2006***
SALARIES	*57,0*	*74,8*	*110,5*
FUEL	*75,8*	*77,8*	*101,7*
SERVICES	*14,3*	*21,5*	*29,2*
SPARE PARTS AND CONSUMABLE SUPPLIES	*13,7*	*13,8*	*24,0*
OTHERS	*10,5*	*11,5*	*17,2*
ROYALTIES	*4,4*	*3,9*	*5,6*
SCHEDULED SHUTDOWNS	*21,5*	*23,5*	*45,4*
AMORTIZATION AND INTERESTS	*0,0*	*0,0*	*0,0*
TOTAL	***197,1***	***226,8***	***333,6***

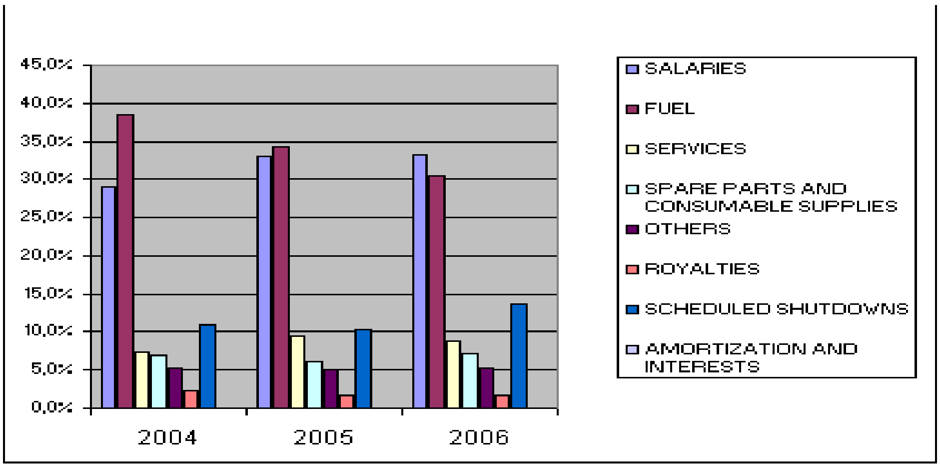


Figure 3.11.5 Evolution of O&M costs (excluding taxe)s

3.11.3. COMPLIANCE WITH THE OBLIGATIONS IMPOSED BY THE CONVENTION

In spite of the economic difficulties Argentina faced, the nuclear electric generation maintained its quality level both in the security and availability of the power stations, meeting to all the regulatory requirements.

The preceding considerations reflex that the Licensee had 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 clearly distinguished processes: the analysis of incidents and the global and systematic study 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 correct human errors, identifying the deficiencies regarding organisation, 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 accidental situations).

These pre-accidental and post-accidental errors are analysed in the same way as the behaviour of components, equipment and systems, 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.

Human corrective actions carried out at the NPPs as a result of PSAs were the following:

3.12.1.1. HUMAN CORRECTIVE ACTIONS IN CNE

- *Emergency water supply system*
  - *Emergency Procedure modification to improve the operator reliability to provide lake water supply in case of refill dousing water,.*
  - *System test procedure modifications,*
  - *Operator action modification due to procedure mistakes.*
- *Emergency Core Cooling System*
  - *Emergency Procedure modification for recovery actions of specific components,.*
  - *System test procedure modifications to avoid misalignments,*
  - *Some components were included within the surveillance program,*
  - *Operator action Improvement related to high/low pressure manual connection,*
  - *Changes from manual to automatic actions of specific operator actions to improve reliability,*
  - *Operator action modification due to procedure mistakes.*
- *Service Water System*
  - *Emergency Procedure modification for recovery actions to supply electric power in case of loss of service water,*
  - *Better procedures to improve the operator reliability to consider the Emergency water supply system in case of loss of service water.*
- *Feedwater system*
  - *Link improvements among emergency operating procedures.*
- *Moderator system*
  - *Shutdown test improvements.*
- *Electrical system.*
  - *Maintenance improvements for specific components,*
  - *Test improvements.*

3.12.1.2. HUMAN CORRECTIVE ACTIONS IN CNA I

Many human action improvements were described in previous reports. The backfitting program included a large number of design changes and procedure modifications. After the backfitting 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 as a consequence new procedures were carried out improving the overall plant safety. The size of the backfitting program implied the review of all plant procedures and the critical human actions that combined with the PSA results give rise to improve its reliability.

The latest procedure changes were the following:

- *Emergency procedure modifications and update such as "loss of feedwater system", "loss of house-pumps" and "loss of off-site power",*
- *Additional test was included regarding the second heat sink (emergency feedwater 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 organisational deficiencies, therefore these events may be used as leading indicators in anticipating and preventing declining performance.
- Detection of Organisational (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 systematisation of the evaluation process involves the use of international applicable methodologies to evaluate human performance in CNA I and CNE.

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. Assessment of human performance can be made using items like:

- *Pre-job briefings,*
- *Training plans, including plant abnormalities,*
- *Safety indicators such as collective dose and surveillance deviations,*
- *Use of procedures and instructions,*
- *Organizational and individual Safety Culture.*

The following weaknesses were identified and the corresponding corrective actions were implemented:

- *Procedures: lack of completeness, lack of clear acceptance criteria, mistakes and poor human action reliability. Corrective action: Changes and update in operating procedures.*
- *Training: lack of specific training, inadequate training. Corrective action: changes and update in personnel training and retraining.*
- *Plant Systems: inadequate design and poor ergonomic design. Corrective action: upgrading of systems and components.*
- *Additionally, other results of applying different methodologies were found such as: inadequate communication, lack of planning, lack of supervision and lack of resources. Corrective action: Changes in the corresponding management policies.*

During the past 3 years 19 events were reported by the Licensee. Of those, about 7 events with root and contributing causes related with human factors have been found. Some of them were the main cause of the occurrence of the event. 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 OPEX have been used to avoid occurrence or recurrence of events.

Some examples where the analysis were completed (included lessons learned) are as follow (see details in Annex VIII):

- *CNA I: Unforeseen personnel exposure,*
- *CNE: Refuelling machine heavy water spilling A,*
- *CNE: Refuelling machine heavy water spilling C,*
- *CNE: Primary heat transport system - liquid relief valve spurious opening.*

Events screening and analysis by technical working groups of the ARN are useful tools when it is necessary to request:

- *Training or re-training of plant personnel,*
- *Updating of procedures,*
- *Generation of changes in managerial attitude towards safety culture,*
- *Investigation of detected precursors of significant events.*

Lessons learned and corrective actions are followed up by inspections and regulatory audits. Moreover, considering the experience gathered during the CNA I and CNE PSA, the periodic training of CNE personnel at Gentilly-II simulator in Canada, CNA I personnel at Angra simulator in Brazil, as well as the human reliability analyses carried out for CNE and CNA I 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 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, standards AR 3.2.1 and AR 3.4.1 establish the information the operator should count with in order to take safety related decisions, the prohibition of interventions during the period immediately after the occurrence of accident initiating events and the characteristics of the man-machine interaction related to the design of the reactor instrumentation and protection systems.

Another aspect to take into account in the prevention of human errors are the measures taken by the Licensee related with the contractors in order to ensure a sufficient competence and safety culture. In that sense 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.

3.12.2. 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, supported by the Licensee, 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, through human reliability analysis application, 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 foreign simulators of compatible plants, as there are no NPP full scope simulators in the country. 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 in the PSAs results,*
- *Fostering the training program addressing past wrong 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 analyzed 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 programmes 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 QA programs in the design, construction, commissioning, operation and decommissioning stages of nuclear installations is a regulatory requirement in Argentina. In the case of NPPs, standard AR 3.6.1 sets the requirements to be fulfilled. In addition, standard AR 3.7.1 determines when the Licensee must present the QA program and 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 or these specific programs is the General Quality Assurance Program of the Licensee.

QA programs and manuals, among other documents, are also mandatory for the installation.

The ARN controls NPPs Quality Assurance Programs implemented by the Licensee by audits carried out according to the usual methodology.

3.13.2. QUALITY ASSURANCE PROGRAM OF THE LICENSEE

Since NASA's creation, the convenience arose of having a general QA Program that would be the reference frame for the specific quality programs of each organizational unit.

The Quality Assurance General Manual describes the Quality System. 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 NASA, 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 IX). CNA I and CNE develop their own programs according to the requirements set in the General Manual.

NASA'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.

Quality Management at NASA was reorganized in March 2005. Although the former organization of the QA area had worked satisfactorily, the experience achieved over the past years has demonstrated the convenience of modifying the dependence of the QA area of the operating stations, in order to optimize the available resources and to obtain a better coordination and integration of common objectives and policies.

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 the Company.

Periodically, the QA Management issues reports showing the audits' results. These reports are then sent to NASA'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 new chart organization for QA Management.

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

ORGANISATION UNIT	***DOCUMENT***	***NUMBER OF PROCEDURES***
NASA	*Quality Assurance General Manual - Approved July 2004*	*18*
CNA I	*Quality Assurance Manual for the Operation of CNA I - Approved December 2005*	*260 (April-07)*
CNE	*Quality Assurance Manual for the Operation of CNE – Approved April 2007*	*638 (Feb 07)*
CNA II	*Quality Assurance Manual for the Construction of CNA II – Approved February 2007*	*132 (June 07)*
Engineering and Support Services	*Services Department - Quality Assurance Manual*	*154 (June 07)*
Engineering and Support Services	*Engineering Department - Quality Assurance Manual*	*14*

QA Programs for NASA, 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.

The QA General Manual is currently under revision, in order to adequate it to the Management System for Facilities and Activities – Safety Requirements GS-R-3, IAEA, 2006.

The QA Manual for the construction of CNA II was upgraded following IAEA Practice Code 50-C-Q, ISO Standard 9001:2000 and Regulatory Standard: AR 3.6.1. This upgrading comprises occupational Health and Safety requirements, as well as GS-R-3, IAEA, 2006, Occupational Health and Safety - OHSAS 18000 (IRAM 3800-Safety and Occupational Health) and ISO Standard 14000 Environmental Management System for Environmental Management.

Annex IX shows NASAS's Quality Policy and NASA's Environmental Policy.

The Licensee verifies that every person and every organization involved in the nuclear area becomes thoroughly familiar with QA requirements. By using 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 introduced or changes are 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. Besides, their performance is mainly evaluated when carrying out tasks related to programmed maintenance tasks. The Licensee's staff supervises tasks carried out by temporary personnel.

NASA has 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 from CNE, the maintenance, erection and commissioning of CNA II and the technical and administrative management of the Central Office. The Environmental Management System was first certified in August 18, 2003; and was renewed in August 12, 2006.

Changes performed in the QA organization have allowed a better coordination of the audits performed from the Central Offices and from each station. 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.

Figures 3.13.2, 3.13.3, 3.13.4 and 3.13.5 show NASA, CNA I, CNE and CNA II Organizational Chart respectively.

The most important changes in the Organizational charts are the creation of the Life Extension Management area, and the creation of the Atucha II NPP Management Unit for the erection and commissioning of CNA II.

ARN audits the QA programs of the Licensee central offices and of the nuclear installations 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 such subject issued by the ARN.

3.13.3. COMPLIANCE WITH THE OBLIGATIONS IMPOSED BY THE CONVENTION

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

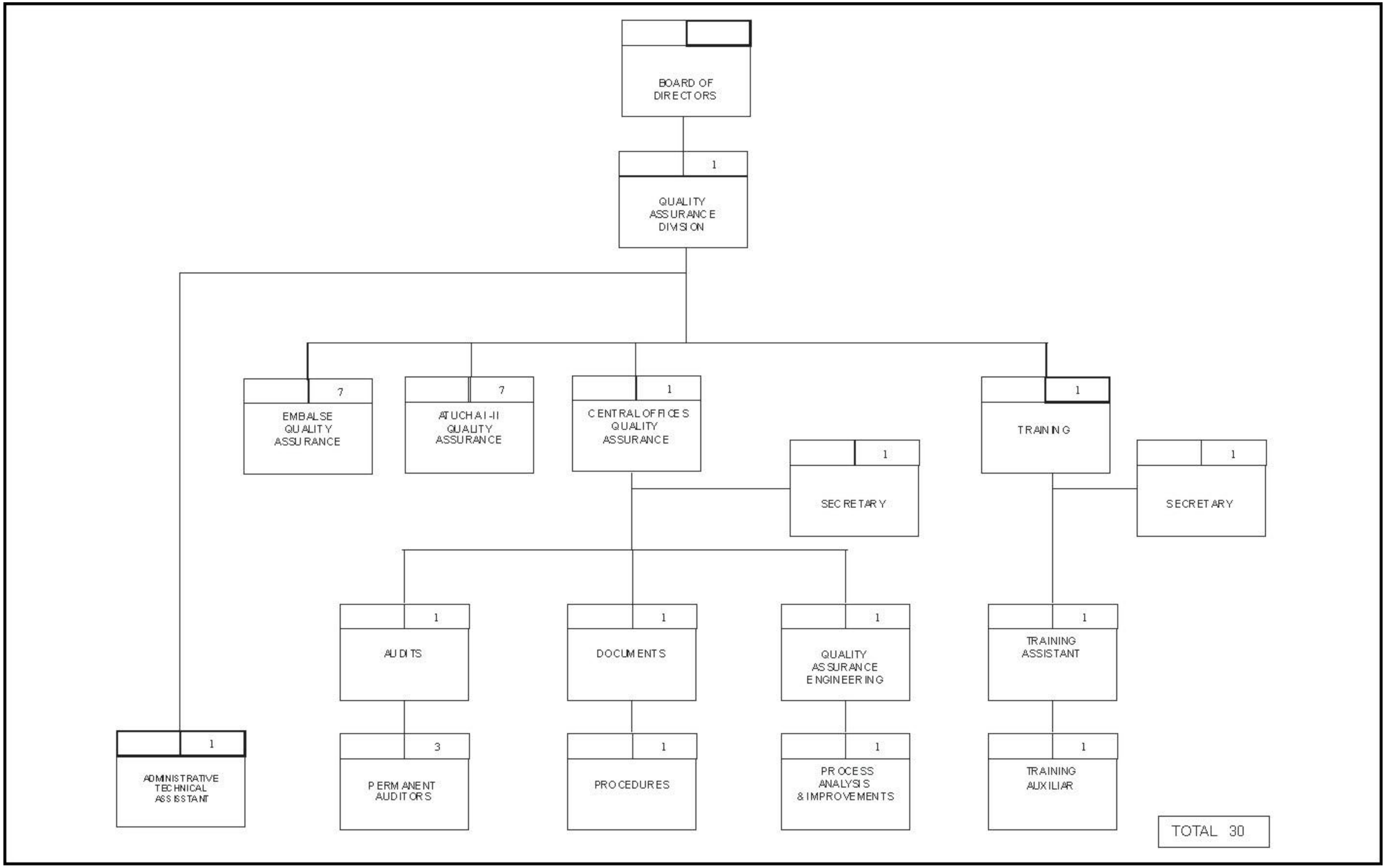


Figure 3.13.1 - Quality Assurance Division

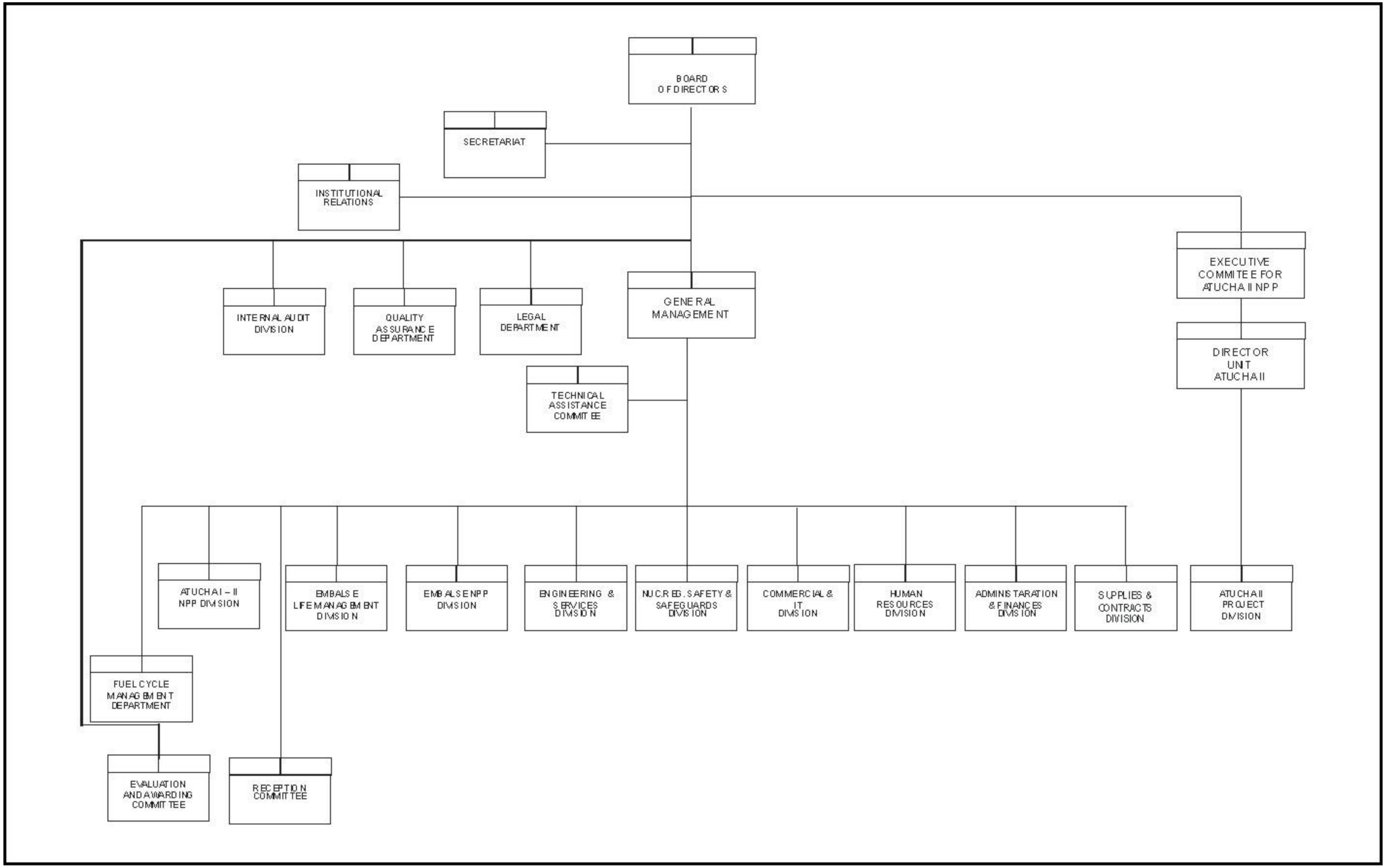


Figure 3.13.2 - Nucleoeléctrica Argentina S.A. General Organisation Chart

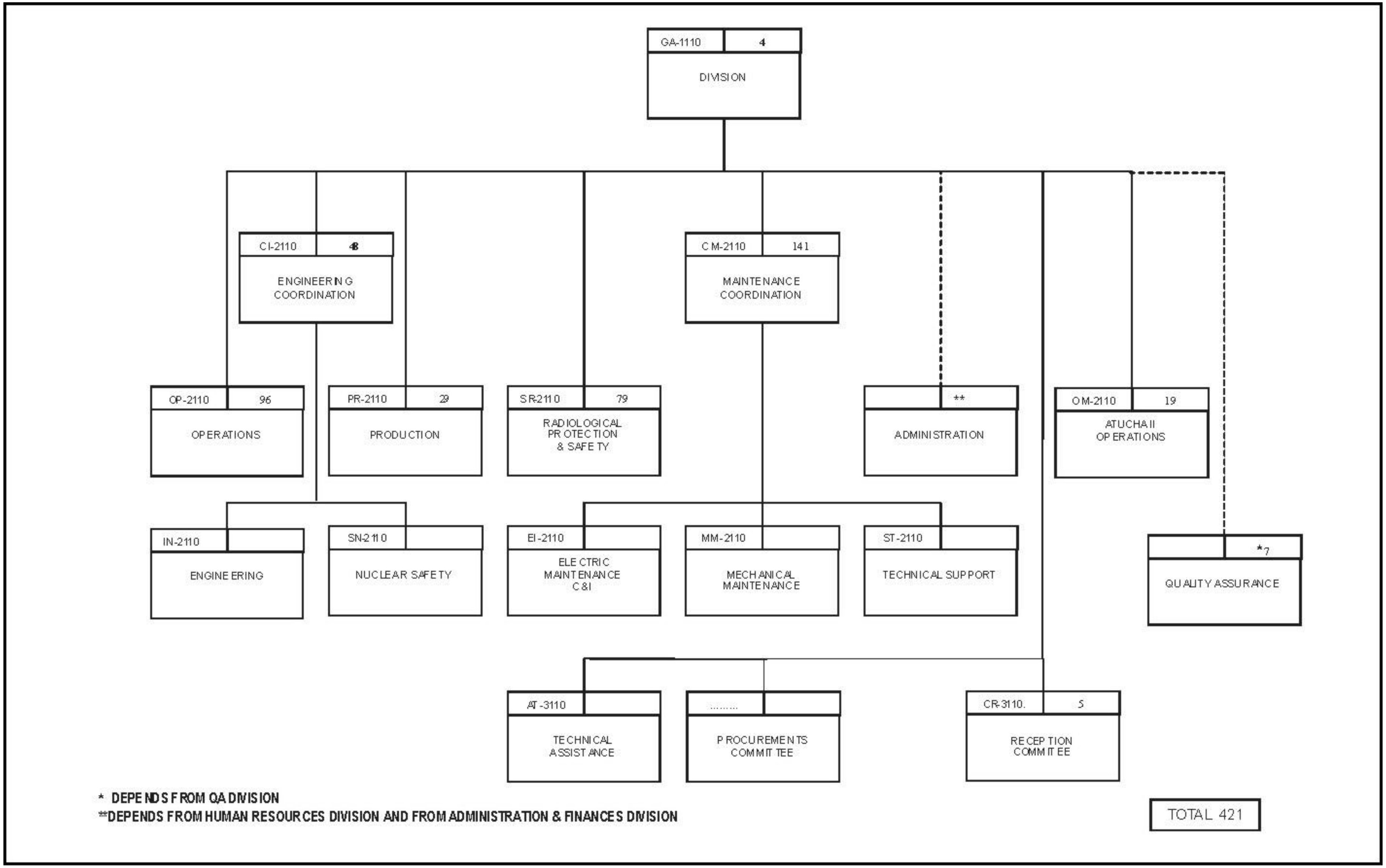


Figure 3.13.3 - Atucha I NPP Chart

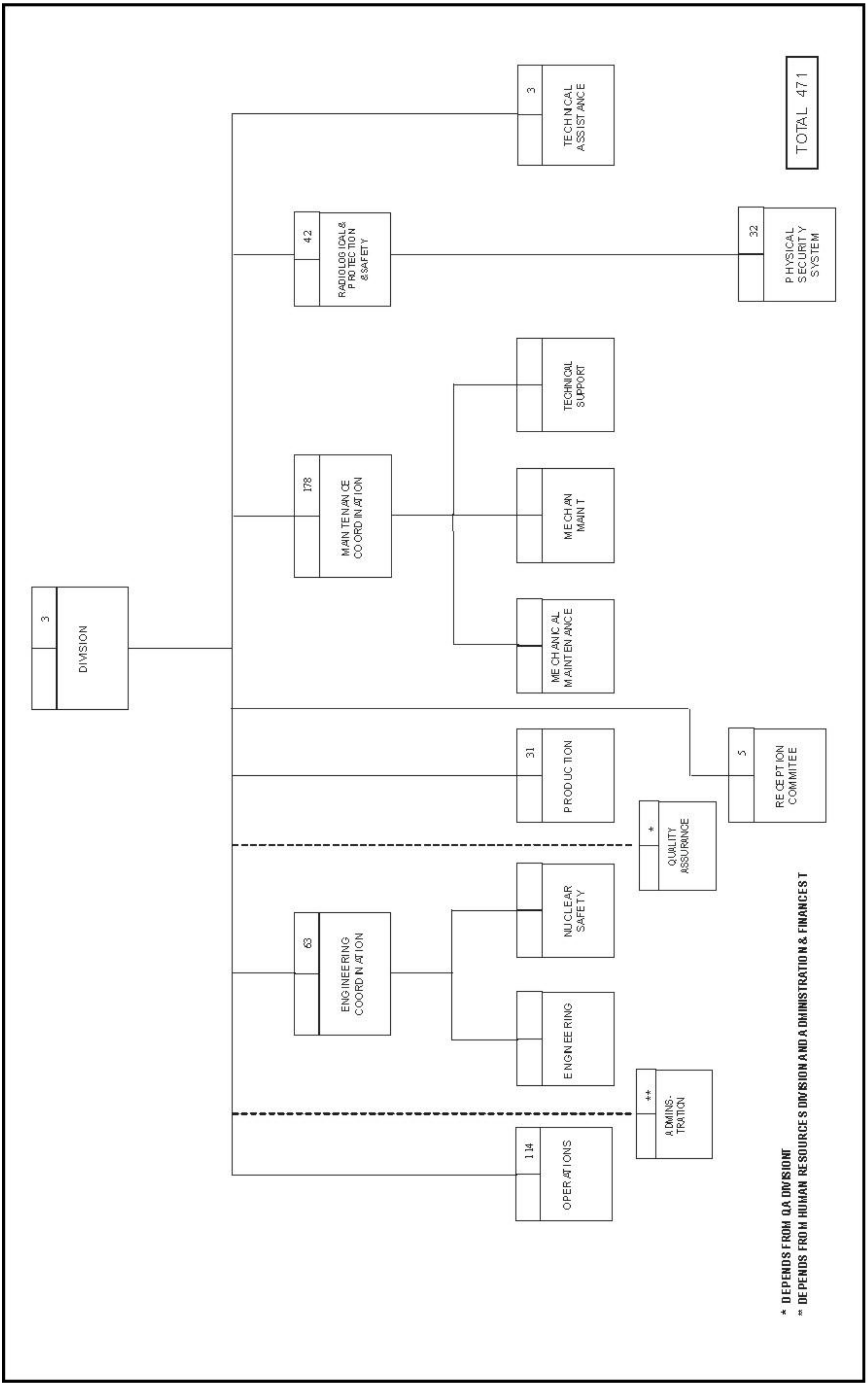


Figure 3.13.4 - Embalse NPP Chart

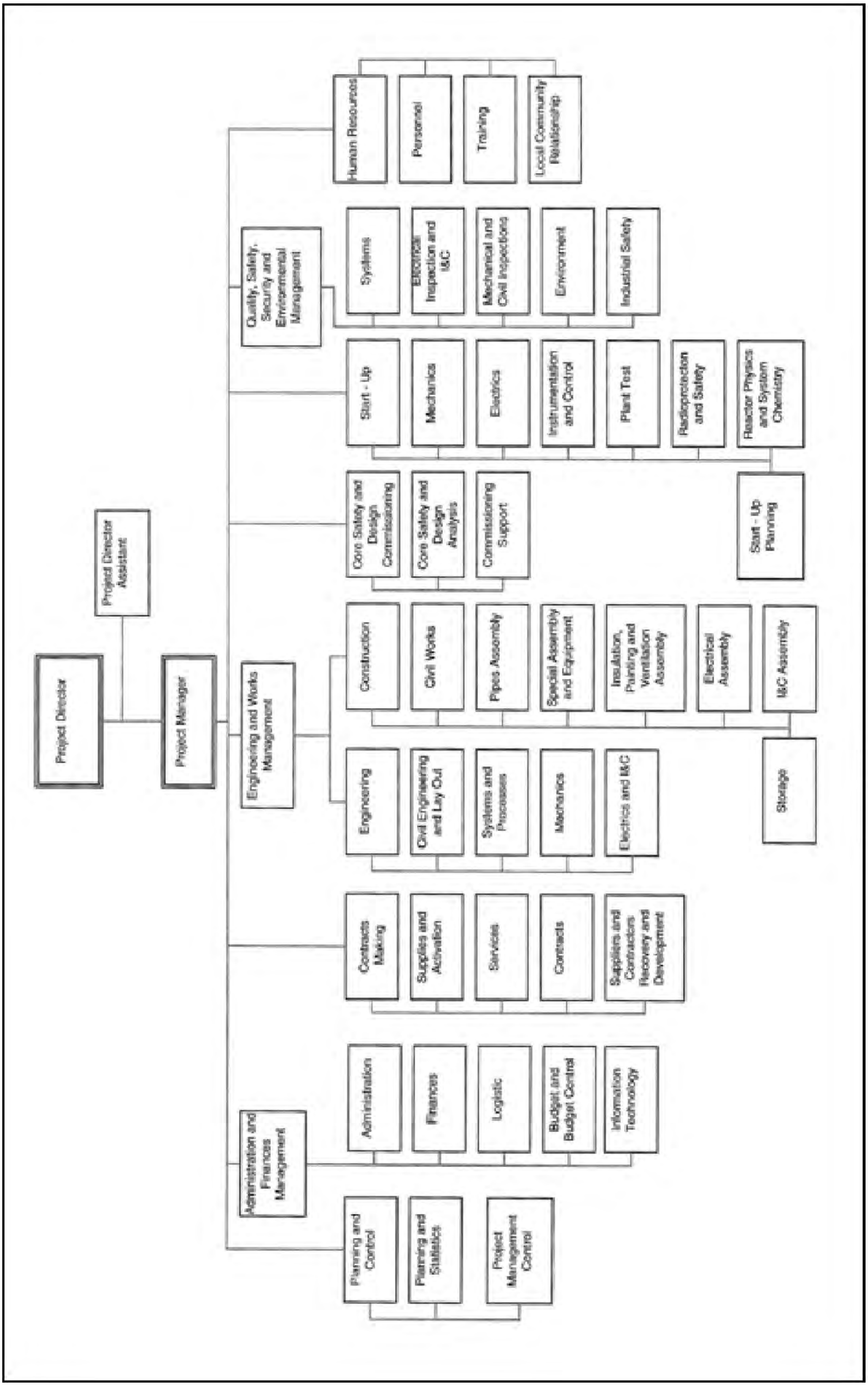


Figure 3.13.5 - Atucha II NPP Chart

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 a ARN requirement or as demand of the Licensee itself. ARN controls the safety level by means of 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 safety assessments above mentioned involve the periodic revision of the probable failure modes of structures, components and systems, 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 such 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 Licence.

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. Such 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 Chapter 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. Such 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 2004 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, and ageing monitoring, prevention and mitigation.

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 Program to analyze 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. These program consists 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.

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.

Ageing assessment of CNE is given in Subsection 3.14.3.1.3.

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 (about 33 years in the case of CNA I), 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 to the analysis of operational incidents and accidents that may occur in this plant or any other one, 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 Convention are shown in subsections 3.14.3.1 and 3.14.3.2.

3.14.3.1. DETERMINISTIC ASSESSMENTS

3.14.3.1.1. 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 steal equivalent to DIN 22NiMoCr37 and similar to ASTM A 508 class 2 forging.

Initially, KWU (the Designer), didn't consider 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 design life time (EOL- end of life, corresponding to 32 years of full power operation). Nevertheless, when during further evaluations the neutron fluence was found to be able to reach higher values, 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.

The program consisted in irradiating samples of the RPV's most critical base material as well as their respective welding, in the lower part of some of the CNA I cooling channels. The differences in the neutronic spectrums in these positions respect of the RPV's wall, led to an important uncertainty in the evaluation of the results, which implied not been able to use the results in a safety study.

The first valid irradiation tests within the irradiation program in host reactors were undertaken by KWU in 1983 in the German VAK reactor. They were done on irradiated and also non-irradiated samples and consisted of various kinds of tests: tensile, Charpy impact, drop weight and fracture toughness. 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 advise of KWU/Siemens, experts and international consultants.

An important milestone which should be taken into account concerning the RPV's material, was the technical meeting which took place at CNA I in the beginning of 2002, with the participation of international experts from various technical scientific institutions (Oak Ridge National Lab., GRS, VTT-Technical Research Centre of Finland, and FRAMATOME ANP) and experts from NASA, CNEA and ARN. During this meeting everything done up to that moment related to the RPV material properties was discussed and further studies were proposed, which were all undertaken.

These additional studies comprised additional irradiation programs in research reactors with neutron spectrums and irradiation temperatures similar to those of the CNA I RPV. The research reactors that were used for the irradiations were the BR2 in Belgium (2003) and the Loviisa in Finland (2003 - 2005). In these occasions the samples underwent diverse tests at internationally acknowledged laboratories.

In June 2006, NASA submitted to ARN a report done by an international consultant lead by CEN/SCK of Belgium. The report presents the complete and detailed analysis of all the data and results obtained from the CNA I RPV surveillance program, including the tests of the irradiated samples at BR2 and Loviisa, as well as presenting a safety evaluation of the RPV for EOL and for 1.5 times EOL.

The report shows the re-evaluation done on the 87 data samples that contain the 770 test results using the internationally developed latest tools and a safety evaluation. For the safety evaluation, the acceptance criteria defined in the US NRG 1.99.2 regulatory guide, were applied.

The report also gives numerous results on irradiations done on similar material to that of CNA I's RPV, which support the hypothesis of the absence of the "flux effect" or "lead factor" in the accelerated irradiations done on material from CNA I in the above mentioned research reactors.

From the conclusions should be highlighted the demonstration of the consistency of the results of all the tests undertaken in the host irradiation programs and a model that explains the uncertainties from those corresponding to the surveillance program (samples irradiated at the CNA I RPV).

The report concludes 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.

Also could be mentioned that in addition to the tests and evaluations done to the RPV´s material, non-destructive test were undertaken on the primary welding of the RPV and on the heat affected zone, as part of the In Service Inspection Program, whose results show that so far no relevant flaw indications have been found .

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 end of life. 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.

3.14.3.1.2. Accident Management and Severe Accident Management Program

As was mentioned in the 2004 national convention report, in September 2003 the ARN required the Licensee the development of a Severe Accident Management Program (SAMP) for both plants in operation, starting with CNA I. The activities covered during 2003 – 2004 in SAMP for CNA I and communicated to the third nuclear Convention were:

- *Development of internal procedures for the program (Organization, QA , etc),*
- *Selection of a methodology to characterize plant damage scenarios,*
- *Selection of six plant damage state scenarios to be characterized during the first stage of the program,*
- *Starting the development of a model for severe accident progression. MELCOR package was selected.*

The activities covered during 2004-2006 in SAMP for CNA I are those related to plant damage states characterization, identification of new scenarios, grouping of accident sequences, proposal of strategies issuing of reports corresponding to findings coming from better knowledge of plant behaviour, and improving what was used in previous PSA analysis. Those tasks are closely followed by ARN personnel. It could be mentioned:

1. *Plant damage states characterization*

   This task includes the development of new deterministic studies (or extending the existing ones) with RELAP code up to the initiation of core uncoverage.

2. *Strategies proposal (preventive strategies for the time).*
   - *Small LOCA with failure of low pressure emergency water injection.*
   - *Small LOCA plus moderator pumps failure which delays low pressure emergency water injection because of depressurization delay.*
     - *This is possible only for a particular break size.*
     - *The strategy proposed uses the volume control system pumps to feed water into the primary. At the same time they are fed through a pump that takes water from a demineralized light water reservoir.*
     - *The strategy proved to be effective as a preventive measure for breaks up to 20 cm^2 in any primary circuit location.*
   - *Blackout*
     - *The strategy comprises two manual actions. One is to avoid air entering the primary, which deteriorates Secondary Heat Sink (SHS) action (through SGs). The other is to avoid, or at least significantly reduce, the probability of LOCA scenarios.*
     - *A Shift Supervisor is working with the SAMP group in order to identify the design changes necessary to incorporate the strategy.*
   - *Unavailability of the second heat sink pumps or lack of sufficient inventory for different accidental scenarios.*
     - *The strategy applies to all scenarios in which SGs are the available heat sink and SHS (emergency water supply system) tank or pumps are partially or totally unavailable. The objective is to extend SHS mission time by adding water to the SHS tank, or directly to the SGs, in order to have more time to recover the plant safety equipment necessary to keep the plant in a controlled shutdown. The strategy takes into consideration equipment that is already in the plant.*

   At present a new strategy is analyzed related to the use of an alternative control room in case the main control room becomes uninhabitable. The postulated scenario is fire in the main control room.

Also under analysis is the possibility of venting the containment considering different alternatives of opening the sphere, taking into account the experience in other German design containments.

3. *Progress in the development of the accident progression model*

   This task was started by a group from Cuyo University (CEDIAC) and it was later continued by the Division of Nuclear Safety from CNEA. The work performed by this second group were focused mainly on the revision of the previous model in order to improve it, the revision of input plant data and the development of specific models for reactor coolant system, moderator, pressurizer and steam generators. A single channel model was also developed in order to better analyze the behavior of the fuel element and its channel during core damage.

   The results of these analysis were presented at the meetings: Cooperative Severe Accident Research Program (CSARP), and at the MELCOR users, where the peculiarities of CNA I were discussed with other specialists that attended the conference.

   The CNEA group worked in this project until October 2006 when it was transferred to CNA II PSA level 2. The experience gained in the CNA I modeling is being used to develop the CNA II model with MELCOR. With GRS assistance, the same group will continue with the severe accident plant progression. It is expected that in a few months this experience will be transferred back to SAMP for CNA I.

4. *Other issues related with severe accidents*

   Some years ago, the Regulatory Body started to study different issues related with severe accidents such as containment failure modes of CNA I, hydrogen behavior and associated mitigation systems, with the purpose to establish the basis for potential requirements. Currently, as a new model of severe accident progression is being developed with MELCOR for CNA I, those former studies are no longer used.

   Besides, as was pointed out in the third versions of this report, the design of CNA II considers hydrogen mitigation devices. The design of these devices is currently being evaluated. The experience gained through these devices design will be used as a basis for future CNA I evaluations.

5. *Severe Accident Management team definition*

   With the collaboration of CNA I personnel, 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.

In connection with CNE accident progression analysis, the one carried out for "Generic CANDU Probabilistic Safety Assessment" and the "CANDU 6 Probabilistic Safety Study" developed by AECL is considered, and a number of core damage accidental sequences were selected as results from the PSA level 1 developed. On the other hand, the Licensee is maintaining close discussions with AECL to develop an international methodology for analysis of accident progression in CANDU reactors.

3.14.3.1.3. CNE Life Extension

NASA has begun a CNE life extension feasibility study in order to establish the scope, schedule and necessary investments for the project.

Therefore, NASA has decided to undertake, with the support of AECL, a Plant Life Management (PLIM) and a Plant Life Extension (PLEX) program in order to achieve a long term operation, and to maintain a high level of safety and plant performance. 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.

The PLEX of CNE is divided into three phases, as follows:

Phase 1: Screening of SSC´s, Ageing Assessment, Safety Assessment, Design Change Packages, Economic Assessment and Licensing.

Phase 2: Retubing Pre-Project, Engineering and material supply for retubing, Specification and detail engineering for other system and components, Materials supply.

Phase 3: Retubing, Implementation of general modifications.

At present, NASA is working on the tasks mentioned in Phase 1, and the retubing related tasks of Phase 2. The scope and the state of the tasks to be performed in CNE are the following:

- *Retubing: This methodology and the process itself consists, as minimum in the removal of feeders, pressure tubes and calandria tubes in a planned manner according to the defined logistics. The retubing pre-project activities includes:*
    - *Review of the reactor and Service Building and the surrounding plant structures to determine the capabilities and limitations related to retube logistics.*
    - *Site specific dimensional and radiation field measurements.*
    - *A conceptual design of a permanent Waste Storage Facilities and waste characterization due to the retubing process.*
- *Ageing Assessment*

  Plant ageing assessment, both Condition Assessment and Life Assessment, is a part of an integrated strategy to assess the active and passive ageing degradation of the CNE components, and then to assess the prognosis for service life extension. Also maintenance and inspection requirements could be determined as well as the necessary upgrades or replacement needed to achieve the life extension in safe condition.

  The Ageing Assessment activities that have been carried out are: Screening and prioritisation of SSCs; Ageing Assessment pilot studies and some modifications or corrective actions are in execution or have already been performed. An important effort is being made in this area, with the incorporation to this subject of an important number of professionals, due to its importance for the decision making relative to the analyzed components state and life extension.
- *Safety Assessment*

  In the pre-project activities a detailed review is undertaken of the safety and licensing issues. The objective of the Safety Assessment is to determine the areas of the plant susceptible to improvements.

  At the present, Plant Safety Assessment for CNE Life Extension, includes: Periodic Safety Review according to the IAEA safety guide NS-G-2.10; review and update of Deterministic Safety Analysis; PSA Level 2; assessment of the Licensing Basis; review of Design Changes in other CANDU plants; review of CNE design against regulatory standards; review and update of Hazard Assessments and review of Trip Coverage.
- *Digital Control Computer Replacement*

  Although the Digital Control Computer used in CNE have had a good performance, the replacement is required for long term operation because of reasons related to obsolescence, maintenance personnel, aging and possibility of expansion. In order to deal with the replacement, CNE participates in a COG Joint Project.
- *PLEX Activities Performed in 2007 Planned Outage*

  The following activities were planned responding to specific objectives of the life extension program: repair works on piping supports and concrete structures according to the findings of the walkdowns performed during 2005 planned outage; repair works on containment structure; walkdowns on Balance of Plant and NP areas, seismic walkdown, and continuation of retubing pre-project tasks that had begun in the 2005 planned outage.

3.14.3.1.4. 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.

3.14.3.1.4.1. 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 contact between these and the CT.

The tasks are carried out during the planned outages, and consist on repositioning of the garters springs of a group of fuel channels according to a priority, in order to:

- *avoid the contact of the PT with the corresponding CT, before their EOL,*
- *eliminating existing contacts, and*

- *to verify, in case that the 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.*

Up to now, the garter springs in 344 PT (90% of the whole number) have been repositioned and 36 PT (10% of the whole number) remain to be intervened.

NASA uses for repositioning, the technical criteria based on the variation of the deuterium up-take 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 over the rate of deuterium incorporation.

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 at the tool for Garter Spring liberation. NASA conserve a close relationship with AECL engineering and continue incorporating all the operating experience of the Canadian plants, reaching a better estimation about the time window for contact between PT and CT.

3.14.3.1.4.2. Pressure Tube Inspection and Scrapping

Following the program established for long-term assessment of hydride blister susceptibility by monitoring deuterium uptake in PT at CNE, a new scraping and ISI of ten PT, were carried out during the last planned outage in April-May 2007, with the following scope:

- *Taking samples by scrapping of the 10 PT, at different longitudinal positions. The channel selection for scrapping was done in accordance with deuterium up-take figures determined in previous campaigns (May of 2004 and Oct 1998), and was the result of considering the following conditions: PT analyzed previously, manufactured ingots of PT not analyzed until the present, PT with the highest deuterium increment, PT with high content of initial hydrogen, channels left in contact to less than 230.000 EFPH and fuel channels with loss of water toward annulus gas system. The samples were sent to the laboratories of AECL in Canada for analysis. The results are still not available.*
- *In-service Inspection. The selection of the channels to be inspected was carried out to fulfill the requirements of the CAN/CSA N285.4-94 standard. The approach to select the channels kept in mind the factory records, assembly, operation and previous inspections. The inspection consisted on carrying out:*
  - *Detection of flaws by ultrasound inspections,*
  - *Detection of flaws by eddy currents inspection,*
  - *Garters spring locations,*
  - *Gap measurement between the PT and CT,*
  - *Measurements of wall thickness and diameter of the PT,*
  - *Sag measurements of the fuel channel,*
  - *Length measurement of the PT.*

  New mechanisms of degradation were not found during the ISI of the 10 fuel channels of the PT; indications were not found with characteristic of sharp flaws; and the deformation of the PT are equivalent to the other PT of CANDU 6 reactors.

  A problem of humidity in the annular gas system, due to the ingress of light water coming from the lattice tube of the channel V08, was solved. During 2004 planned outage, the annular gas system was modified in order to isolate the channels line containing channel V-08. Additionally, the Licensee took actions and demonstrated and assured the Regulatory Body, that safety is not compromised with the annular gas system modification. In the mean time, AECL is studying the method to repair or give a solution for this leak.

3.14.3.1.5. 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 of feeders are inspected, with the objective of determining the wall thickness reduction produced by flow-accelerated corrosion (FAC), and crack detection at feeder bends.

The inspection was 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 of 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 complain with the crack detection program.

During the planned outage 2007, 141 feeders were inspected. The conclusions of previous planned outages related to a thinness rate, due to FAC, slightly lower than other CANDU reactors, were confirmed by the measurement results, while there were no clues of relevant indications of flaws . The main contributor to these performance is the chemical control to the primary heavy water. The results in the evolution of the inspection that took place allow a life expectancy up to the EOL for all sensitive areas without the necessity of replacements.

3.14.3.2. PROBABILISTIC SAFETY ASSESSMENT

The activities covered during 2004 – 2006 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, has been developed to an enhanced level 1 according to the objectives of the back-fitting program. That program had added new safety systems, such as the Second Heat Sink and redundancy improvements as in the secondary relief valve system. No new modifications of the design have been analyzed since the time of the last convention meeting.

The enhanced PSA for CNA I is undergoing a revision, by the Licensee, on the internal events subject. This matter is aimed at a Core Damage Frequency (CDF) result more realistic than the latter, by adding the outcomes of more recent thermo-hydraulic studies that are being performed in the frame of the Severe Accident Management Program. Conservative hypothesis adopted in former studies can be ruled out with new ones in force. E.g., small LOCA through the pressurizer relief valve along with the Low Pressure Safety Injection failure, used to be the accident with the largest contribution to the CDF. Such accident sequence does not lead to a core damage during the mission time, according to the most recent studies.

In February 2005, NASA submitted a technical evaluation of a time between planned outages, from 12 to 18 months, to the ARN. NASA evaluated the impact on the nuclear safety when frequencies of preventive maintenance, routine tests and in service inspection programs are to be modified. 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 $1{,}3 \times 10^{-5}$ /year.

A specific acceptance criterion for a CDF increment due to design changes is not currently included in the argentine standards. The US-NRC recommendation on this subject sets a limit to the increase value of CDF due to design modifications. This limit is set 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.

At present an update of CNA I PSA is ongoing, taking into account new findings coming from a better knowledge of the plant behavior as was mentioned before, and CDF frequency is expected to be reduced.

3.14.3.2.2. CNE Probabilistic Safety Assessment applications

A PSA with corrective actions was accomplished by the utility since 2004 until now. The ARN asked for such actions to be included within the operation procedures and the re-training program of the personnel.

The scope of the PSA spanned to the low power state. That yielded an 8.8×10^{-6}/year figure, which is 35% of the full power one.

A qualitative study on other potential sources of radioactive release (Spent Fuel Bay, Cobalt rods, spent fuel elements dry storage, etc) was also carried out by NASA as an improvement of the study.

The first figures of the PSA study were obtained by using a free code known as PSAPACK (developed by IAEA). For the sake of a second review the model underwent a migration to the Risk Spectrum code.

Regarding other PSA applications, it can be mentioned that the impact of the “mejillón dorado” bivalves, which clogs the heat exchanger and reduces its efficiency, was evaluated. NASA as a requirement of the ARN did this study. It was based basically upon considering some affected cooling *system, such as the Service Water system and Emergency Water Supply system. It was found that the Emergency Core Cooling System (ECCS) was the most impaired system, since it has only one heat exchanger that exchanges heat to the secondary side. This component was tagged as "non repairable" and a service inspection document was issued: a bi-monthly surveillance flow measurement. Such criterion was applied to other heat exchangers, cooled by service water. A surveillance of those components is also part of the preventive maintenance program. Analyzing the expected changes in the CDF figure was adopted as a main criterion in this case.*

Another application of the PSA was the extended time between outages from 12 to 18 months. At this point it has to be mentioned that a time period outages close to 18 months is a common practice in most of CANDU reactors around the world.

The ARN requirement was to evaluate the impact on the nuclear safety when frequencies of preventive maintenance, routine tests and in service inspection programs are to be reduced.

As a part of the requirement, NASA developed a study using the PSA level 1 tool with the plant at full power. Such study accomplished the fulfillments established by the US-NRC 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.

3.14.4. REGULATORY PLANT SAFETY PERFORMANCE INDICATORS

From 1998 performance safety indicator data were collected and evaluated as was explained in the previous national convention reports. Until 2002, the indicators evaluation was made throughout the analysis of changes on their behaviour, but there were no acceptability criteria.

In order to establish thresholds or acceptability values for performance indicators, it was necessary to analyze historical data, but getting historical data was not possible for most of the indicators at the beginning of the program. Statistics was made for those indicators that have been reported in the past (outages, power reductions, dose, training, wastes and effluents), but such a method was not applicable for indicators in areas like maintenance or repetitive tests.

In 2002, frequency distributions of each indicator were made using the data collected since 1998. From those distributions an acceptability criteria was defined and a pilot implementation was initiated for validation. As a result of the pilot implementation experience, evaluation criteria were changed.

Thresholds for indicators of each NPP were calculated separately because plant performances are not comparable.

Validation or modification of the defined limits is a continuous task. Some indicators have an almost constant value along time and good operational conditions are observed so this value could be considered as an acceptable reference value.

However, it is difficult to define an optimum, acceptable or unacceptable value. Even for indicators for which regulatory limits are applicable, they cannot be used as a threshold because the historical values of the indicators are below those limits.

From 2006 and taking into account its own experience plus IAEA documentation, the number of safety performance indicators used in ARN was modified to 24. The set in current use is the following:

Normal Operation

- ***Plant Stability:***
  - *01. Number of plant shutdowns.*
  - *02. Number of power reductions.*
  - *03. Load Factor:*
- ***Radiological Protection***

  Dose:
  - *04. Individual maximum dose.*
  - *05. Total equivalent dose.*

  Effluents:
  - *06. Liquid effluent discharges.*
  - *07. Gaseous effluent discharges.*

Waste Management:

08. Low activity solid wastes.

- ***Surveillance:***

Maintenance:

09. Number of reports on safety or safety-related system deficiencies of a corrective nature submitted during the period.

10. Number of reports on safety or safety-related system deficiencies of a corrective nature which are still pending, excluding those requiring the cooling down of the nuclear power plant.

11. Number of reports on safety or safety-related system deficiencies of a corrective nature which are still pending due to lack of supplies.

12. 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. Number of overdue preventive or predictive routine inspections and maintenance tasks involving safety or safety-related system components, excluding those requiring the cooling down of the power plant.

Repetitive Tests:

14. Number of overdue repetitive tests of safety or safety-related systems.

15. Number of deficiency reports submitted on the basis of the repetitive tests performed on safety or safety-related systems.

16. Number of test procedures whose revision or issuance is overdue.

- ***Organization***

Training:

17. Number of hours devoted to training on safety-related issues.

Feedback from Operational Experience:

18. Number of documented event analyses, findings or design modifications in similar power plants.

Internal Control:

19. Number of internal technical audits.

Compliance with Regulatory Authority standards

20. Number of pending Regulatory Requirements.

Abnormal Operation

- ***Events***

21. Number of relevant events.

22. Number of activation of the Safety Systems.

- ***Risk***

23. Unavailability of Safety Systems (under revision due to the fact that it is considered necessary to clarify its definition)

24. Impact of reported events on core damage frequency (under revision due to the fact that it is considered necessary to clarify its definition).

It should be mentioned that Argentinean experience in the use of Safety Performance Indicators showed 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 an important tool in order 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 but not using it on its own but together with other tools, such as event analysis, audits, inspections, (among others), for monitoring safety.

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 Argentina NPPs in operation, demonstrated 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

There have not been any changes in the general criteria and standards used in Radiological Protection in Argentina, in the last period.

The control and supervision of the compliance with standards and other regulatory documents are completed 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 different working groups belonging to the ARN, who perform analyses and evaluations related to different topics on radiological safety. These working teams have their own laboratories so that they are able to perform the measurements and experiments required for this purpose.

These controls are performed routinely, but they are especially carried out when it is necessary to reinforce the inspection tasks, as in the case of programmed outages and non-foreseen shutdowns or as a consequence of some specific situation.

The periodic test program related to radiological protection is monitored and observed during its performance, experimental data coming out from the mentioned tests are confirmed. Among this set of tests, those related to radiation detection equipment installed in different working areas and the execution of emergency plan implementation exercises stand out.

The personnel dosimetry system is evaluated for external irradiation and for internal contamination by means of specific audits carried out by ARN specialists and requiring the participation of dosimetry laboratories in intercomparison exercises.

Regarding the control of effluents released to the environment by the installations, the present measurement plan during the annual operation consists of the measurement of the released activity in those places where effluents are emitted. This plan consists of a planned routine measuring timetable and is complemented with random controls.

In addition to the environmental monitoring plan carried out by the installations, the ARN independently performs environmental measurements in the surroundings of CNA I and CNE and nearby zones with its own laboratories and specialists. The set of control points, where samples for this plan are taken, include not only those selected by the installation but also some other points chosen according to ARN criterion.

3.15.2. ATUCHA I NUCLEAR POWER PLANT

3.15.2.1. RADIOACTIVE RELEASES INTO THE ENVIRONMENT

The authorised limits to the environmental releases of CNA I were slightly modified in January 2004, due to the update in dose assessment model and parameters.

Table 3.15.1. - Authorised gaseous discharge limits for CNA I

NUCLEID	K_i (TBq)
Ar-41	1×10^{3}
Ba-140	4×10^{0}
Co-60	8×10^{-2}
Cs-134	1×10^{-1}
Cs-137	6×10^{-2}
H-3	1×10^{4}
I-131	5×10^{-2}
Kr-85m	1×10^{4}

NUCLEID	K_i (TBq)
Kr-88	1×10^{3}
Ru-103	1×10^{0}
Ru-106	7×10^{-2}
Sb-122	3×10^{1}
Sb-124	2×10^{0}
Sr-89	5×10^{-1}
Sr-90	6×10^{-3}
Xe-133	6×10^{4}
Xe-135	8×10^{3}

Table 3.15.2. - Authorised liquid discharge limits for CNA I

NUCLEID	K_i (TBq)
Ag-110m	5×10^{1}
Ba-140	1×10^{2}
Ce-144	4×10^{1}
Co-58	2×10^{2}
Co-60	8×10^{0}
Cr-51	8×10^{3}
Cs-134	2×10^{0}
Cs-137	3×10^{0}
Fe-59	8×10^{1}
H-3	2×10^{4}
I-131	1×10^{1}

NUCLEID	K_i (TBq)
Mn-54	1×10^{2}
Ni-65	4×10^{3}
Ru-103	4×10^{2}
Ru-106	3×10^{1}
Sb-122	2×10^{2}
Sb-124	1×10^{2}
Sb-125	8×10^{1}
Sr-89	9×10^{1}
Sr-90	1×10^{1}
Transuranics	4×10^{0}
Zn-65	2×10^{1}
Zr-95	1×10^{2}

The gaseous radioactive releases to the environment due to CNA I operation in the period 2004-2006 may be observed in Table 3.15.3, discriminating those corresponding to I-131, tritium aerosols and noble gases; it also includes an estimation of C-14 discharged.

Table 3.15.3. - 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)
2004	$1,5 \times 10^{-4}$	$5,5 \times 10^{2}$	$5,4 \times 10^{-6}$	$1,8 \times 10^{2}$	$5,6 \times 10^{-1}$
2005	$2,0 \times 10^{-4}$	$1,4 \times 10^{3}$	$7,0 \times 10^{-6}$	$8,8 \times 10^{2}$	$4,1 \times 10^{-1}$
2006	$1,6 \times 10^{-4}$	$6,2 \times 10^{2}$	$5,3 \times 10^{-6}$	$1,3 \times 10^{2}$	$4,3 \times 10^{-1}$

In 2005 the increase in the discharge of noble gases was due to faults in the fuel elements and the increase in the tritium discharges were due to a defect in the nipple of channel NO4 (see the events in Annex VIII).

The liquid radioactive releases to the environment by CNA I during the same period are presented in Table 3.15.4, discriminating between liquid discharges of tritium and other radionuclides.

Table 3.15.4. - Activity released from CNA I to the environment as liquid discharges

YEAR	TRITIUM (TBq)	OTHER RADIONUCLIDES (TBq)
2004	$1,1 \times 10^{3}$	$2,6 \times 10^{-1}$
2005	$1,4 \times 10^{3}$	$2,6 \times 10^{-1}$
2006	$1,2 \times 10^{3}$	$2,5 \times 10^{-1}$

In 2005, the increases in the tritium discharges were due to the event: "Lowering of tank TR41B001´s volume level" (see the event in Annex VIII).

Of the total annual average discharges from CNA I to the environment, 84% corresponds to tritium. Comparing these annual average discharges with the respective annual authorised discharge limits, we observe that they were less than 9% of these limits.

CNA I measures and reports the radioactive releases 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 location, habits and food consumption and local dispersion factors of environmental releases. Same considerations are valid for CNE.

3.15.2.2. PUBLIC EXPOSURE

The annual average dose to the critical group, due to CNA I operation during the period 2004-2006, was lower than 2% of the established individual dose constraint, being gaseous discharges the main contributor. The annual dose values are shown in Table 3.15.5, discriminated according to discharge type.

Table 3.15.5. - Critical group individual dose for CNA I

YEAR	***GASEOUS DISCHARGE DOSES (mSv)***	***LIQUID DISCHARGE DOSES (mSv)***	***TOTAL DOSES (mSv)***
2004	*$2{,}9 \times 10^{-3}$*	*$1{,}2 \times 10^{-3}$*	*$4{,}1 \times 10^{-3}$*
2005	*$6{,}6 \times 10^{-3}$*	*$1{,}1 \times 10^{-3}$*	*$7{,}7 \times 10^{-3}$*
2006	*$2{,}9 \times 10^{-3}$*	*$1{,}2 \times 10^{-3}$*	*$4{,}1 \times 10^{-3}$*

The annual values of collective effective dose normalised per unit of electric energy generated $(GW_{(e)}\ y)^{-1}$, are presented in Table 3.15.6, calculated over population data up to a radius of 2000 km from the NPP.

Table 3.15.6. - Regional normalised collective effective dose for CNA I

YEAR	***GASEOUS DISCHARGE DOSES (man Sv/Gwa)***	***LIQUID DISCHARGE DOSES (man Sv/Gwa)***	***TOTAL DOSES (man Sv/Gwa)***
2004	*$2{,}7 \times 10^{-1}$*	*$7{,}6 \times 10^{-1}$*	*$1{,}0 \times 10^{0}$*
2005	*$1{,}0 \times 10^{0}$*	*$1{,}3 \times 10^{0}$*	*$2{,}3 \times 10^{0}$*
2006	*$3{,}9 \times 10^{-1}$*	*$1{,}1 \times 10^{0}$*	*$1{,}5 \times 10^{0}$*

The increase in the discharge of noble gases during 2005, explained in preceding paragraphs, provoked an increase in the collective dose and in that of the critical group.

The annual average collective effective dose per unit of electric energy generated, for the period 2004-2006, represented 11% of the collective effective dose constraint per unit of electric energy generated, set by ARN in 15 [man Sv $(GW_{(e)}\ y)^{-1}$], for NPPs design.

Furthermore, the global annual average collective effective dose per unit of electric energy generated, due to tritium releases was 1,6 [man Sv $(GW_{(e)}\ y)^{-1}$] for the period 2004-2006.

3.15.3. EMBALSE NUCLEAR POWER PLANT

3.15.3.1. RADIOACTIVE RELEASES INTO THE ENVIRONMENT

The Regulatory Body authorised a set of gaseous and liquid discharge limits, which correspond to a critical group dose much lower than 0.3 mSv. They are shown in Tables 3.15.7 and 3.15.8, respectively.

Table 3.15.7. - Authorised gaseous discharge limits for CNE

NUCLEID	**K_i (TBq)**
Ar-41	$7{,}4 \times 10^{3}$
Kr-85m	$3{,}7 \times 10^{4}$
Kr-87	$7{,}4 \times 10^{3}$
Kr-88	$3{,}7 \times 10^{3}$
Xe-133	$1{,}9 \times 10^{5}$
Xe-135	$3{,}7 \times 10^{4}$
H-3	$3{,}7 \times 10^{4}$
I-131	$2{,}2 \times 10^{1}$

NUCLEID	**K_i (TBq)**
Co-58	$3{,}7 \times 10^{1}$
Co-60	$3{,}7 \times 10^{-1}$
Sr-89	$1{,}1 \times 10^{2}$
Sr-90	$3{,}7 \times 10^{0}$
Ru-106	$1{,}5 \times 10^{0}$
Cs-134	$1{,}5 \times 10^{0}$
Cs-137	$3{,}7 \times 10^{-1}$
Ba-140	$1{,}5 \times 10^{2}$

Table 3.15.8. - Authorised liquid discharge limits for CNE

NUCLEID	K_i (TBq)
H-3	$3,7 \times 10^{3}$
Cr-51	$3,7 \times 10^{2}$
Mn-54	$7,4 \times 10^{-1}$
Fe-59	$3,7 \times 10^{1}$
Co-60	$1,5 \times 10^{-1}$
Zn-65	$7,4 \times 10^{-2}$
Ni-65	$7,4 \times 10^{3}$
Sr-89	$3,7 \times 10^{0}$
Sr-90	$1,5 \times 10^{-1}$
Zr-95	$1,9 \times 10^{0}$

NUCLEID	K_i (TBq)
Ru-103	$3,7 \times 10^{0}$
Ru-106	$1,5 \times 10^{-1}$
Ag-110m	$1,1 \times 10^{0}$
Sb-125	$1,1 \times 10^{0}$
I-131	$1,9 \times 10^{-1}$
Cs-134	$3,7 \times 10^{-2}$
Cs-137	$3,7 \times 10^{-2}$
Ba-140	$1,1 \times 10^{1}$
Ce-144	$1,9 \times 10^{-1}$
Gd-153	$3,0 \times 10^{1}$

The gaseous radioactive releases by CNE to the environment, for the period 2004-2006 may be seen in Table 3.15.9, discriminating those corresponding to I-131, tritium, aerosols and noble gases and including an estimation of C-14 discharges.

Table 3.15.9. - Activity released from CNE to the environment as gaseous discharges

YEAR	***I - 131 (TBq)***	***TRITIUM (TBq)***	***AEROSOLS (TBq)***	***NOBLE GASES (TBq)***	***C -14 (TBq)***
2004	$5,3 \times 10^{-7}$	$3,3 \times 10^{2}$	$8,3 \times 10^{-7}$	$4,5 \times 10^{1}$	$4,4 \times 10^{-1}$
2005	$1,2 \times 10^{-8}$	$3,4 \times 10^{2}$	$3,8 \times 10^{-7}$	$3,4 \times 10^{1}$	$4,2 \times 10^{-1}$
2006	$3,2 \times 10^{-7}$	$4,0 \times 10^{2}$	$5,1 \times 10^{-7}$	$3,5 \times 10^{1}$	$4,8 \times 10^{-1}$

The liquid radioactive discharges to the environment by CNE, for the same period, are presented in Table 3.15.10, discriminating between liquid discharges of tritium and other radionuclides.

Table 3.15.10. - Activity released from CNE to the environment as liquid discharges

YEAR	***TRITIUM (TBq)***	***OTHER RADIONUCLIDES (TBq)***
2004	$8,3 \times 10^{1}$	$1,9 \times 10^{-3}$
2005	$7,0 \times 10^{1}$	$3,5 \times 10^{-3}$
2006	$1,6 \times 10^{2}$	$7,5 \times 10^{-3}$

Of the total annual average discharges from CNE to the environment, 92% corresponds to tritium. These annual average discharges were less than 3% of the respective annual authorised discharge limits.

3.15.3.2. PUBLIC EXPOSURE

The annual critical group doses due to CNE´s operation during the period 2004-2006 are presented in Table 15.11, discriminated according to discharge type. The annual average value resulted lower than 2% of the established individual dose constraint. The liquid discharges were the main contributor.

Table 3.15.11. - Critical group individual dose for CNE

YEAR	***GASEOUS DISCHARGE DOSES (mSv)***	***LIQUID DISCHARGE DOSES (mSv)***	***TOTAL DOSES (mSv)***
2004	$1,9 \times 10^{-4}$	$2,3 \times 10^{-3}$	$2,5 \times 10^{-3}$
2005	$2,2 \times 10^{-4}$	$2,5 \times 10^{-3}$	$2,7 \times 10^{-3}$
2006	$2,2 \times 10^{-4}$	$4,6 \times 10^{-3}$	$4,8 \times 10^{-3}$

The mayor 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 collective effective dose normalised per unit of electric energy generated is presented in Table 3.15.12, calculated over population data up to a radius of 2000 km from the NPP.

Table 3.15.12. - Regional normalised collective effective dose for CNE

YEAR	GASEOUS DISCHARGE DOSES (man Sv/GWa)	LIQUID DISCHARGE DOSES (man Sv/GWa)	TOTAL DOSES (man Sv/GWa)
2004	$1,8 \times 10^{-2}$	$8,3 \times 10^{-2}$	$1,0 \times 10^{-1}$
2005	$1,9 \times 10^{-2}$	$8,0 \times 10^{-2}$	$9,9 \times 10^{-2}$
2006	$1,9 \times 10^{-2}$	$1,5 \times 10^{-1}$	$1,7 \times 10^{-1}$

The annual average collective effective dose per unit of electric energy generated, for the period 2004-2006, represented less than 1% of the collective effective dose constraint per unit of electric energy generated, set by the ARN for NPPs design.

The global annual average collective effective dose per unit of electric energy generated due to tritium releases was 0.2 [man Sv $(GW_{(e)}\ y)^{-1}$] for in the period 2004-2006

3.15.4. OCCUPATIONAL EXPOSURE

The radiological protection criteria used by the Regulatory Body to control the dose received by workers are consistent with the latest ICRP recommendations.

AR 3.1.1 regulatory standard sets different criteria to ensure that the occupational doses to workers are as low as reasonably achievable and lower than the established dose constraints, and that the protection is optimised.

ARN requires that whenever possible, radiological protection be achieved using installation systems rather than operational procedures.

Each NPP´s Operating License sets the following conditions for workers:

- Personnel working in a controlled area must be submitted to individual monitoring and annual medical surveillance.
- Monthly occupational doses due to external exposure, and intake of radioactive material in this period, must be recorded.

  These records must contain the following information:

  - Individual dose.
  - Collective effective dose 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

According to what standard AR 10.1.1 establishes, dose limits have not been exceeded when the following conditions are fulfilled:

$$\frac{H_p(d)}{L_{DT}} \leq 1$$

and

$$\frac{H_p(10)}{20\,\mathrm{mSV}} + \sum_j \frac{I_j}{I_{L,j}} \leq 1$$

Where:

$H_p(d)$ is the personnel equivalent dose at a depth of 0.07 mm and 3 mm (for skin and crystalline respectively), integrated in a year,

L_{DT} is the 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 doses, the normalised collective effective doses and the average individual effective doses received by workers in CNA I during the period 2004-2006, are presented in Table 15.13.

Table 3.15.13. - Occupational Dose in CNA I

YEAR	COLLECTIVE EFFECTIVE DOSES (man Sv)	NORMALIZED COLLECTIVE EFFECTIVE DOSES (man Sv/GWa)	AVERAGE EFFECTIVE DOSES (mSv)
2004	1,9	6	4
2005	6,4	26	6
2006	9,3	36	8

During 2004 there weren't any programmed outages at CNA I, which explains the lower dose value compared to the following years.

In 2005 an internal incident of radiological characteristics took place at CNA I, where a worker received an individual dose 10% above the dose limit. The Operating Organisation decided that this worker should no longer undertake tasks in the controlled area until the time when he were in condition to do so. Additionally, intensive re-training plans were implemented for all the personnel.

3.15.4.1.2. Occupational dose at CNE

The collective effective doses, the normalised collective effective doses and the average individual effective doses received by CNE workers during the period 2004-2006 are presented in Table 3.15.14.

Table 3.15.14. - Occupational Dose in CNE

YEAR	COLLECTIVE EFFECTIVE DOSES (man Sv)	NORMALIZED COLLECTIVE EFFECTIVE DOSES (man Sv/GWa)	AVERAGE EFFECTIVE DOSES (mSv)
2004	3,0	5	3
2005	2,9	5	3
2006	0,5	1	1

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.

During 2006 there was not any programmed outages at CNE, which explains the lower dose value compared to the preceding years.

3.15.5. ALARA ACTIVITIES

ALARA program is carried out in both NPPs during normal operation and during outages. Each NPP have 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.

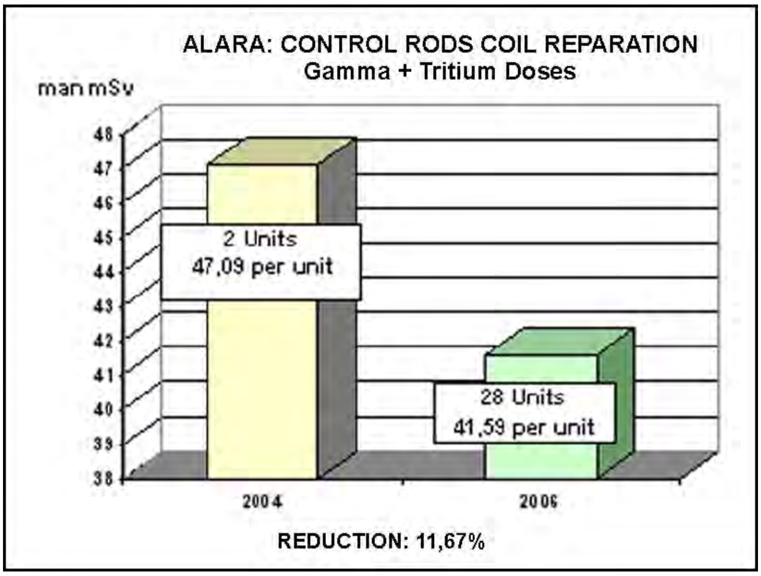
The ALARA activities have been reinforced and intensified with the objective of optimizing the dose received by personnel occupationally exposed to ionizing radiations, reinforcing its activity with better implementation of mock-ups, training, improvement in tools and the analysis of work plans.

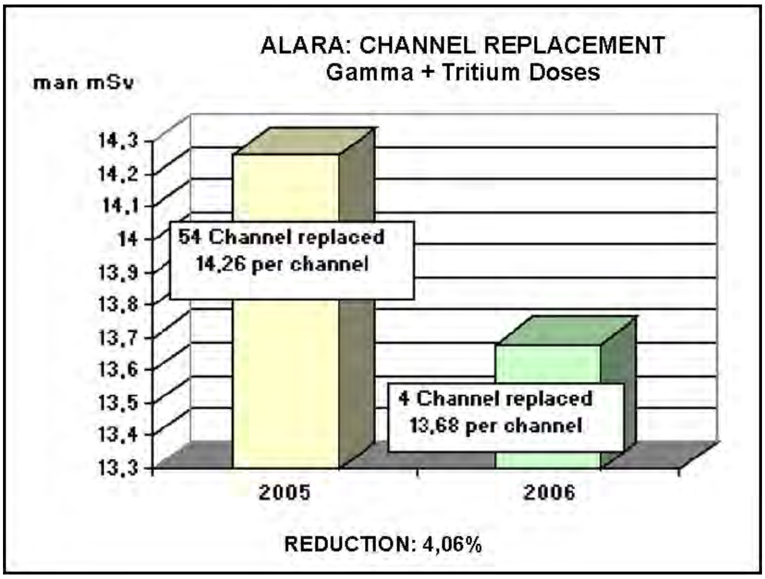
Some of the main ALARA improvements made during NPP outages, for high exposure activities was related to control rod, steam generators inspection, main coolant pumps seal replacement, channel replacement, etc.

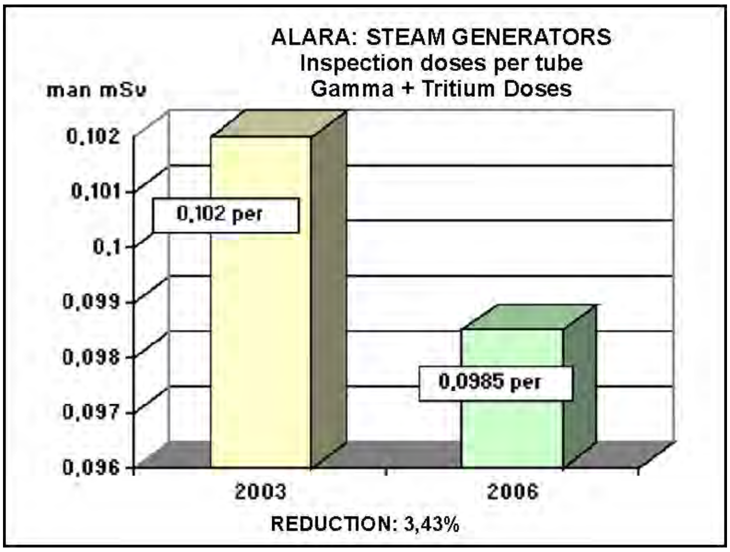
The following specific issues are examples:

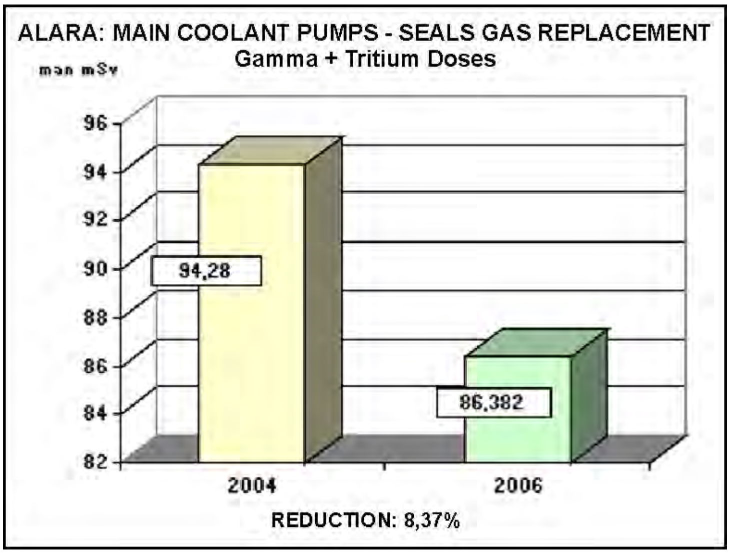
- *Shielding: for control rod coil reparations, main coolant pumps seal replacement, steam generator inspections and channel replacements, the shields were improved and for moderator heat exchanger reparation an adequate shield was implemented.*
- *Training: specific training programs were developed for the personnel involved in tasks like shielding, control rod coil disassemblies and mock-ups related to steam generators.*
- *Tools: new extracting tools for fuel channels and control rod tubes were implemented, as well as changing working methods for welding, improved methodology related to the removal and transport operations with channels and control rod tube guides.*

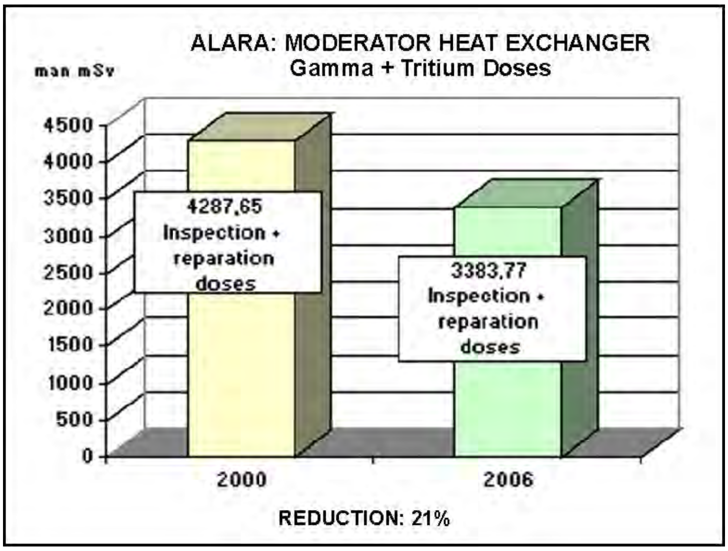
The following charts show the improvements in many relevant areas:











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.

Therefore the country complies with the obligations imposed in Article 16 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 on-site 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 radiological emergency not only inside but also outside NPPs. Such plan, usually known as "Internal and External Emergency Plan", shall comprise every aspect related to the strategy required to control and limit the accident’s consequences. This plan is part of the documentation required by the ARN to grant the license of operation.

In November 1998, the National Executive Power subscribed Decree 1390 defining the scope and procedures facilitating the enforcement of the Act No 24,804, or “National Law of Nuclear Activities”.

The application of Decree 1390 caused a change in the ARN functions, powers and duties related to radiological offsite aspects concerning nuclear emergencies preparedness, training and response.

The main important issues were:

- The ARN is empowered to regulate and control nuclear activity as well as advise the Executive on issues under its purview, including radiological and nuclear emergencies.
- The ARN must provide protection against harmful effects of ionising radiation even under emergency situations.
- The ARN shall set the guidelines and criteria for the emergency plans and training of members of neighbouring public in case of nuclear accidents.
- Emergency plans shall be developed by local, state and national authorities; they must contemplate an active participation of the entire community and be approved by the ARN.
- The ARN shall conduct the actions within the area covered by the emergencies plans. Security Forces and Representatives of Civil Institutions shall report to the designated ARN officer.

These changes and their consequences on the organizations will be detailed in section 3.16.2.

On the other hand and as an explicit demand contained in the Operating License, emergency exercises are annually carried out, aiming at the evaluation of the participating groups response and the improvement of the plan. The specific objectives of each annual exercise must be agreed with the ARN.

The main objectives of the emergency exercises are:

- to extend the diffusion of the current procedures among the intervening organizations,
- to establish operational commands and, to verify the capacity of application of the automatic radiological precautionary measures (residential sheltering, thyroid prophylaxis, control of accesses, notifications, messages to the population, etc),
- To involve the members of the public wherever actions are planned in detail (up to 10 kilometres) and to practice them.

3.16.2. ARN FUNCTIONS RELATED TO RADIOLOGICAL EMERGENCIES

The ARN continues dealing with prevention of accidents at nuclear installations as its regulatory function.

The criteria that must be adopted by the Licensee were set in the ARN document called “Criteria for the intervention in nuclear emergencies with off-site radiological consequences”. Its compliance has been required to CNA I and CNE.

In order to accomplish what is set in Act 24804 and Decree 1390, the ARN- Nuclear Emergency Response System (NERS) was created by ARN Resolution Nº 25/99 in November 1999.

The NERS is the organizational scheme that the ARN uses to respond in cases of nuclear emergencies and interact with the national, state and local response organizations (National Emergency Cabinet, States Civil Defence and Local Civil Defence of every Municipality within 10 km around each NPP) to manage effectively nuclear emergencies in preparedness, intervention and recovery stages.

In particular, the operational capacity of the ARN in the management of the implementation of actions to protect the members of the public in the surroundings of the NPPs from the radiological consequences, is currently checked by carrying out emergency exercises.

The ARN, 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 organisations (as established in the Emergency Plan) integrate this command and the ARN coordinates the response teams belonging to civil organisations, (Fire fighter brigades, Civil Defence, etc) security forces (Police, Gendarmerie and Coast Guard) and military institutions (Army, Navy and Air Force). These organisations apply the precautionary measures with their response teams. All these groups have procedures to deal with nuclear emergencies, under ARN coordination.

In order to conduct the actions within the 10 km established as the “precautionary action zone”, a Nuclear Emergency Operative Chief (NEOC) from the ARN is designated and integrated to the Local Emergency Operative Centre (LEOC). The ARN-NEOC shall be the officer to whom civil organizations and security forces report to.

A ARN Emergency Control Centre has been set up at ARN’s Headquarters in order to co-ordinate the NERS. This centre also operates in 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”, as the National Warning Point according to IAEA - Emergency Notification Assistance Technical Operations Manual (ENATOM).

3.16.3. REQUIREMENTS CONCERNING EMERGENCY PLANS

The ARN has regulated the planning and preparedness of responses to emergency situations in the NPPs, through different documents, e.g. 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 emergency plans are, in general, the following:

- The implementation of protective measures of automatic application shall be foreseen within a circular area of a radius of 3 km centred at the installation. Their effective application shall also be planned within a circular sector defined by a central angle of 60° with vertex in the installation, symmetrical with respect to the wind direction and within a radius of 10 km (key-hole).
- Once the accident has occurred, a set of urgent protective measures established in the emergency plan must be taken.

The urgent protective measures carried out on the basis of the plant situation and meteorological conditions without waiting for radioactive measurements in the environment, are basically the following (see Table 3.16.1):

- *Control of access to the emergency zone. The control points are defined in the emergency plan.*
- *Sheltering inside dwelling. This action may be extended for some hours; through the massive communication media, people shall be informed about its end and other instructions regarding the subsequent ventilation of their houses.*
- *Distribution of stable iodine. The distribution of stable iodine (in the shape of potassium iodide tablets) to the involved people shall be implemented. The tablets shall be swallowed one per day, by every person staying within the above mentioned key-hole, including milk-fed babies, children and pregnant women.*

After the accident's first phase has passed and the radioactive material release has ceased, the time to implement the non-urgent protective measures commences.

The implementation of the non-urgent protective measures depends mainly on the measurement results of the radioactive material released to the environment (see Table 3.16.1). The most significant are the following:

- Evacuation from the zones affected by radioactive deposit. This action must be implemented:
  - Compulsorily, in the case that the radiation level from the material deposited on the land reaches or exceeds 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.
- *Intervention in relation to food. The intervention levels adopted by the ARN for the substitution of contaminated foodstuffs for consumption were obtained from an optimising 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, in general, 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 land. Due to its high cost the implementation of this action shall be decided on the basis of an analysis for each specific case.*

Table 3.16.1 - Protective measures and intervention levels

	PROTECTIVE MEASURES	APPLICATION	INTERVENTION LEVELS	COMMENTS
Urgent Protective Measures	Control of access and road block	Always	n.a.(*)	Protective measures carried out on the basis of the plant situation and meteorological conditions.
	Sheltering	Always	n.a. (*)	
	Iodine prophylaxis	Always	n.a. (*)	
Non Urgent Protective Measures	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.
	Re-entering evacuated zones	Always	100 mSv	Dose integrated in one year.
	Land decontamination	Optional	n.a. (*)	This action shall be decided on the basis of a case by case analysis.
	Food ban	Always	Yes	

(*) Non-applicable

Table 3.16.2. - Intervention levels for foodstuff, in Bq/kg, as result of a generic optimisation analysis

Group	Cereals	Vegetables	Green Vegetables	Fruits	Meats	Milk	Fish
1	10	10	15	15	50	10	10
2	1 000	1 000	1 500	1 500	5 000	1 000	5 000

Group 1: ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Am, ^{242}Cm, ^{244}Cm, ^{239}Np

Group 2: ^{137}Cs, ^{134}Cs, ^{131}I, ^{89}Sr, ^{90}Sr, ^{95}Zr, ^{103}Ru, ^{106}Ru, ^{140}Ba, ^{144}Ce

3.16.4. IMPLEMENTATION OF REGULATIONS CONCERNING EMERGENCY PLANS

The basic aspects mentioned in Section 3.16.2 were established not only for the operating NPP's internal and external emergency plans, but also in the agreements celebrated with public organisations. These agreements, which are part of the emergency plans, enable the use of networks, communication systems and equipment belonging to public organisations, as well as fast transportation and their fire fighting resources.

As regards agreements with other organisations, an important example is the treatment of injured, contaminated or irradiated persons. The primary treatment for these cases is foreseen at hospitals located in zones near the NPPs. In case more specialized health centres for the assistance of patients is required, in Buenos Aires City several special agreements have been celebrated with institutions having hospital infrastructure and well-trained personnel for such patients.

3.16.5. CLASSIFICATION OF EMERGENCY SITUATIONS

When an abnormal situation arises at a NPP, the Primary Responsible must: classify the initial emergency situation, apply the plant emergency procedures and notify the external emergency organizations. On- Site and Off-Site emergency classifications are as follows:

On-Site Green Alert: when an abnormal situation is detected that compromises the nuclear safety. The internal emergency procedures are applied and the internal emergency organization is established.

Off-Site Green Alert: Without any delay, Primary Responsible must start implementing urgent preventive precautionary measures to protect public (messages to population, IK pills distribution, preparation to control access and for sheltering, and (as a precautionary measure) evacuation 3 Km around NPP). At the same time, the Primary Responsible requests the formation of the external emergency organization.

On-Site Red Alarm: It is declared by the Primary Responsible when the emission of significant amounts of radioactive material initiated. Internal emergency procedures continue and the situation implies notifying the external emergency organization who immediately declares the Off-Site Red Alarm and applies precautionary measures to protect the public (IK pills ingestion, sheltering, access control, evacuation, consumption restrictions, etc).

The On-site and Off-Site emergency classifications are permanently coordinated and exchange of information between both emergency response centres is guaranteed.

The emergency classification described above is for the local range (10 Km around NPP). At a National level, when the Primary Responsible declares an On-Site Green Alert he must notify the ARN. When the On/Off Site Red Alarm is declared, the ARN declares the Red Alarm at national level too.

For the international notification, the Convention on Early Notification of a Nuclear Accident format is adopted, and the ARN is the National Competent Authority who declares and notifies the alarms to the IAEA and the potentially affected neighbouring countries.

3.16.6. ON-SITE AND OFF-SITE NPP EMERGENCY PLANS

The NPP’s internal and external emergency plans include gathering the necessary information for the planning and management, that both the installations and the public organisations involved in the emergency need, in order to face an accidental situation.

The NPP’s emergency plans are prepared in such a way that the intervention before an accident takes into account mainly the following objectives:

- *To lead the accidental situation,*
- *To assess the potential consequences,*
- *To declare and communicate the corresponding alert states,*
- *To introduce the necessary protective measures in order to avoid or mitigate radiological consequences, or its consequences on the individuals and the environment,*
- *To take the necessary actions to restore the dwelling conditions in the involved zone, at the late stage of the accident.*

The emergency plans of both CNA I and CNE include:

- *Concerning organisation and responsibilities:*
    - *The agreements with public authorities in order to implement the protective measures,*
    - *The responsibilities and functional relationships of the organisations in charge of putting into practice the different protective measures,*
    - *The composition, responsibilities and specific function of the Internal Committee of Emergency Control, its place of meeting and an alternative emergency control room outside the installation.*

- *Concerning the procedures, they include:*
  - *The installation conditions in which the Primary Responsible shall declare the emergency in its different levels:*
    - *Internal alert state inside the installation,*
    - *Alert state outside the site,*
    - *Internal emergency inside the installation,*
    - *Emergency outside the site.*
  - *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,*
    - *Organisation activation in order to face the emergency situation,*
    - *Evaluation of the situation,*
    - *Initiation of protective measures application,*
    - *End of protective measures application,*
    - *Remedial measures.*
  - *The protocols and details of the communication systems necessary to manage the emergency.*
  - *The protective measures to be applied, according to the type of 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 aerosols release.*
  - *The way of implementing protective measures and indicate:*
    - *The circumstances in which protective measures shall be implemented,*
    - *Who will be in charge of their implementation,*
    - *The areas in which precautionary measures must be implemented,*
    - *Circumstances and manner in which the protective measures cancellation shall be decided.*
  - *The protocols for alert communication, information and instructions to the potentially affected population (broadcasting, television, loudspeakers, alarms, etc.).*
  - *The protocols for the control of doses to personnel who act during the emergency and the actions to be taken in case their values exceed the corresponding constraints and prescriptions.*
- *Concerning physical places and equipment:*
  - *Specify the sheltered (under protection) place for the permanent or temporary personnel who perform activities up to a radius of 3 km around the installation,*
  - *Specify the places for gathering personnel in the case a possible evacuation, which shall also be capable of giving eventual sheltering,*
  - *Establish the necessary equipment that must be available for radiological monitoring,*
  - *Establish the places inside as well as outside the installation for the operation of the Internal Committee of Emergency Control, and indicate their characteristics,*
  - *Specify who and where shall inform the massive communication media.*
- *Concerning maintenance of resources, they contain:*
  - *A continuous training program for the NPP staff and for the external organisations participating in the emergency. The program contemplates aspects related to the plan implementation and to general radiological safety,*
  - *A procedure to update general and specific contents of the emergency plan,*
  - *A program for calibration and maintenance of equipment and instrumentation assigned to the performance of tasks during the emergency,*
  - *The expected performance of an annual exercise of the emergency plan application.*

3.16.7. STRUCTURE OF THE EMERGENCY PLAN AT NATIONAL LEVEL

The ARN is a specific National organization that acts in cases of nuclear emergencies as described above.

Nevertheless, organizations involved in nuclear activities keep technical relationship with different national organizations among which, the Interior Security Secretary, Federal Emergency System, Federal Police, Gendarmerie, Naval Prefecture, National Meteorological Service, National Hospitals and National Armed Forces are the most relevant. Many of these national organizations have been co-operating in emergency planning, response and training activities before the Act 24804 was sanctioned.

As was stated in previous National Nuclear Safety Reports, there was no a contingency plan at national level till in 1998 a Project for a "National Civil Protection Program" was established, which included nuclear emergencies and stated the basis for the Federal Emergency System, which is described below.

Towards late 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 state response organizations. The ARN and the CNEA are members of this Federal Emergency System.

In October 1999 the Executive sanctioned Decree 1250 establishing the Federal Emergency System, which does not replace the function of any other response organization at different levels, ensuring that there are no overlapping responsibilities during an accident. However the Federal Emergency System is the highest coordinating structure to which response organizations may turn to regarding preparedness, response and recovery.

The Federal Emergency System, 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 Federal Emergency System staff was concluded and approved by the ARN.

3.16.8. NUCLEAR EMERGENCY PLAN AT MUNICIPALITY LEVEL

The emergency plans of both CNA I and CNE are compatible with those of every Municipality within 10 km around each NPPs. These plans are approved by the ARN and periodically tested by means of drills and exercises in which all the organizations and public involved participate. The exercises performed were satisfactory, considering the objectives proposed.

3.16.9. NUCLEAR EMERGENCY PLAN AT STATE LEVEL

The ARN, the Operator and the Local Civil Defence have continued meeting with the State Civil Defence of those States where NPPs are located (Buenos Aires and Córdoba), and they participate in the exercises. The Nuclear Emergency Plans are also approved by the ARN.

3.16.10. CONTACT AND INFORM TO NEIGHBOUR COUNTRIES IN CASE OF RADIOLOGICAL EMERGENCY

In 1986, Brazil and Argentina signed the Argentinean-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 neighbouring country with NPPs. Argentina has also agreements related to nuclear affairs with Paraguay, Bolivia, Uruguay and Chile.

These agreements facilitate bilateral communications when necessary and, furthermore, 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. Into the framework of those agreements, representatives of Brazil and Uruguay took part as observes in some of the exercises carried out in Argentina

3.16.11. IMPROVEMENT ACTIONS CONCERNING EMERGENCY PREPAREDNESS IN THE ARN

In Argentina a Geographical Information System (GIS) is used for the preparation and response to emergencies. ARN has incorporated information related to the physical, demographic and economic aspects as well as urban infrastructures, institutions and industries to this system.

At present the necessary information is available to use the GIS in the early stage of a nuclear emergency. At this stage, the GIS has the capacity to determine the affected zone and to delimit the zones of application of automatic precautionary measurements (residential sheltering, thyroid prophylaxis, control of accesses, etc.) and to determine the amount of inhabitants affected by the precautionary measurements. At the late stage it also allows to delimit the zones of application of precautionary measurements and the amount of inhabitants affected, which facilitate the environmental monitoring and allows incorporating these measurements to feedback programs for calculation of consequences.

At present ARN has detailed information inside a 10 kilometres radius of the NPPs and less detailed information outside these 10 kilometres. The information for the GIS is continuously updated and extended.

An important modification of the Emergency Control Centre, located at ARN Headquarters, was started in order to improve the operative capacity, to be able to take advantage of the most modern technology and to upgrade the equipment. During the first stage the computer equipment was upgraded; assembling a Local Area Network independent from the network of the Regulatory Authority, with capacity for videoconferencing inside and outside the country (IAEA). These modifications required the improvement of the offices occupied by the Emergency Control Centre.

For the second and third stages it is foreseen to establish a point-to-point connection with the NPPs, to allow the on line transmission of information about the state of plant and environmental measurements in emergency situations, and to permit videoconferencing in a more efficient way. Also, the update of the GIS and the capacity for transmitting information processed at the Emergency Control Centre through Internet with a secure connection is foreseen.

3.16.12. ACTIONS TAKEN FOR THE IMPROVEMENT OF THE EMERGENCY EXERCISES

During 2004-2007 period, it was considered necessary to qualify the emergency response groups from all radiological protection organizations, as well as legal aspects, in which their intervention was framed. The robust 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 measures. To this end all levels of the organizations involved were encouraged to perform and accept the different roles assigned.

The nuclear emergency exercises carried out during the period 2004-2007 were the following:

- *Villa Rumipal and Villa del Dique, CNE 2004 Exercise, October 21 2004*
- *Lima, CNA-I 2005 Exercise, November 15 2005*
- *La Cruz, CNE 2006 Exercise, May 30 2006*
- *Lima, CNA-I 2007 Exercise, May 14 2007*

Besides both Licensee and the ARN, the following organizations participated actively and received the specific training according to that established in the legislation and in the Nuclear Emergency Plans:

- *National Army*
- *National Gendarmerie*
- *Prefecture Naval Argentina*
- *Police of Buenos Aires Province*
- *Police of Córdoba Province*
- *Federal Emergency System*
- *Civil Defence, Córdoba Province*
- *Civil Defence, Buenos Aires Province*
- *Civil Defence, Zárate*
- *Civil Defence, La Cruz*
- *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*

The members of the public in the surroundings of both NPPs also participated (up to 10 kilometres, where actions are planned in detail and it is necessary to practice them). Therefore an important effort in diffusion activities and training was carried out previously to each exercise, with special emphasis on primary and secondary schools of the area. In the exercises of Villa Rumipal and Villa del Dique, CNE 2004 Exercise, 4700 inhabitants participated, in the Lima, CNA-I 2005 Exercise and CNA-I 2007, 8300 inhabitants and in that of La Cruz, CNE 2006 Exercise, 1400 inhabitants.

During all emergency exercises the ability of the ARN to conduct the emergency response teams belonging to all civil organizations and security forces was achieved appropriately.

Furthermore, the personnel of the ARN has acquired the necessary skills to carry out the role of conducting civil organizations and security forces, taking advantage of the operative experience of these organisations. It was also possible to integrate representatives of these organizations to the Emergency Control Centre to conduct the emergency. Notwithstanding, the coordination function assigned to the ARN, each participating organization exercises its own role and competence.

3.16.13. STATUS OF NATIONAL AND INTERNATIONAL AGREEMENTS

In what follows the list of approved nuclear emergency plans, under Law 24804/97, Ordinance 1390/98 and ARN Resolution 25/99 is shown:

- *Municipal Plan for Nuclear Emergencies, La Cruz, Córdoba Province, approved in September 2001.*
- *Municipal Plan for Nuclear Emergencies, Zárate, Buenos Aires Province, approved in January 2002.*
- *National Plan for Nuclear Emergencies, Federal System of Emergencies, approved in December 2002.*
- *Municipal Plan for Nuclear Emergencies, Embalse, Córdoba Province, approved in June 2003.*
- *Provincial Plan of Nuclear Emergency, Ministry of Security, Córdoba Province, approved in November 2003.*
- *Plan of Nuclear Emergencies, External Aspects, Embalse NPP (updated in November 2003).*
- *Plan of Nuclear Emergencies, External Aspects, Atucha I NPP (updated in November 2003).*
- *Municipal Plan for Nuclear Emergencies, Villa Rumipal, Córdoba Province, approved in May 2005.*

Within the quality management system of the ARN, improvements to internal procedures related to emergency preparedness are being continuously performed. Formal and operative activities that have arisen from the Convention of Prompt Notification and Assistance in cases of Radiological Emergencies for Accidents in Nuclear Centres were normally met and responded to. Additionally, Argentina participates in emergency events and in CONVEX International exercises.

In connection with the safety and security of NPPs, emergency preparedness activities have continued. The activities have focused on disseminating information of the radiological risk and the nuclear emergency plans to the population surrounding the NPPs.

3.16.14. 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, as it was highlighted in the conclusions about Argentina during the first review meeting on 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 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 Periodical Safety Review, 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. Such 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 population from accidental situations. These steps were taken into account in the elaboration of the corresponding Emergency Plans.

3.17.2. NORMATIVE ASPECTS

In Argentina, a NPP construction may 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 shall 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 affecting the installation safety.
- Dispersion of radionuclides to the environment, both in normal and accidental conditions.

The Regulatory Body issues such license once the Licensee has demonstrated that the design of the NPP to be built complies with standards and other specific regulatory requirements for the selected site, taking into account the NPP-site interaction.

As was mentioned in Chapter 3.7, the regulatory standard AR-3.10.1 “Earthquake Protection” is being revised taking into account the “state-of-the-art” in the matter and in the process analysis of the following new standards:

- *AR-3.10.2 “General Safety Criteria for Siting”*
- *AR-3.10.3 “Safety Criteria for Evaluation of External Events”*
- *AR-3.10.4 “Criteria for Determining the Potential Effects of the NPP in the Region”*

All these Argentine standards are based on the corresponding IAEA’s safety documents.

3.17.3. EXISTING SITES

Two sites were at the time selected and evaluated as suitable for NPP 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.

Atucha has two independent units, CNA I in operation, and CNA II under construction. At Río Tercero, CNE is presently in operation.

The site studies performed for CNE and CNA I in each case underwent the three following 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 spots 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 plant lifetime.

3.17.4. SITING STUDIES PERFORMED

The original siting studies related to CNA I and CNE location were detailed in the previous Nuclear Safety Convention reports. The most significant external events 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).

With respect to CNA II, due to the fact that the unit is located in the same site as CNA I's, a lot of specific information was available at the time of 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, extreme meteorological phenomena and atmospheric dispersion, and population distribution aspects, as well as to the nuclide transfer mechanism 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 as "Seismic Study of CNA II NPP Siting" reflects the results of such 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.

Another natural external event 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 worth while 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 have also been considered in the plant design basis. In this regard, the effect of an uncontained vapour cloud explosion, which has been taken into account through appropriate layout and structural aspects, was considered.

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.

3.17.5. 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 it necessary to re-evaluate such 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 its 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.

Below there follows a summary of the re-evaluation siting aspects at different times after the original evaluation, as well as those being carried out at present.

3.17.5.1. 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 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).
- Re-evaluation of potential hazards from man-induced events at the plant site, as part of regulatory follow up actions.

3.17.5.2. CNE SITE RE-EVALUATION:

A list of external events 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 event 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 a framework of an integrated, systematic and updated program. Therefore, the plant accomplished the definition, procurement, installation and commissioning of a 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. The most important parameters for system configuration are:*
    - *Trigger or minimum acceleration, corresponding to "Event Detected " signal in control room, and which initiates instrument recording;*

    - *Cumulative Absolute Velocity, which is an indicator of the potential damage due to the earthquake, integrates in time the difference between absolute acceleration and a preset reference value;*
    - *Operative Basis Earthquake (OBE, with an occurrence frequency of 10^{-2} / year; SL-1 according to IAEA SG-50-S1 Rev.1), whose characteristic, if exceeded, trigger the "OBE exceeded" signal in control room.*
- *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 power transmission lines which are essential for the plant safety. A work plan was prepared, starting in December 1998 and 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).*
- *In relation to the re-evaluation of potential hazards from man-induced events at the plant site, the training of experts from the national organization in charge of this activity was continued and a work plan was prepared.*
- *Different alternatives have been studied regarding flooding in CNA I. Actually not only 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 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 storage spent fuel pool. The second heat sink, which is now under construction, will improve safety for these scenarios providing an independent system from the actual residual heat removal system, which uses the river as final heat sink.*

3.17.5.3. RE-EVALUATIONS UNDER EXECUTION

CNE is located in a seismic risk area, and PSA was not considered in original design but many specific studies were made related to seismic risk:

- *Plant responses in case of an earthquake occurrence. Seismic instrumentation.*
- *Fuel Elements load machine seismic responses verification.*
- *Services buildings response in case of an earthquake occurrence.*

These requirements were met by NASA at the right time.

Later on, ARN required the application of the PSA methodology in CNE. The scope included the seismic PSA.

In order to comply with the mentioned regulatory requirement that involves the safety re-evaluation in case of a seismic occurrence at CNE, the Licensee proposed to ARN the application of the Seismic Margin Assessment (SMA) methodology instead of the seismic PSA. That proposal specified the use of the EPRI methodology.

ARN agrees with the NASA proposal with the following conditions:

- *To justify the value to be used for the Review Level Earthquake (RLE) for the CNE site, justifying the intensity and floor spectrum, and the attenuation factors to be used.*
- *Taking into account that EPRI proposes three different scopes, ARN required to use the "total reach" EPRI methodology scope.*

The Licensee is applying the Seismic Margin Assessment (SMA) methodology following ARN conditions.

With respect to this issue, could be mentioned that the internationally accepted and more used methodologies for safety evaluation in case of a seismic occurrence are, the seismic PSA and the SMA methodology. The international experience allows verifying that the conclusions obtained by both methodologies are equivalent as far as the implementation of improvements to NPPs is concerned. However, the time and resources required to perform the re-evaluation are considerable lower in the case of SMA, when compared with seismic PSA.

In CNE original design, only one earthquake level was adopted for the seismic qualification of the safety-related systems, structures and components (SSCs), with a maximum horizontal ground acceleration of 0.15 g and an occurrence frequency of 10-3 /year.

However, before the commissioning, several modifications were performed in SSCs in order to ensure its safety functions in case of occurrence of a maximum horizontal ground seismic of 0.26g that corresponds to an occurrence frequency of 1.4x10-4 /year.

The SMA study began to be performed at CNE applying the methodology developed by EPRI. During the recent planned outage in 2007, a plant walk down made by specialists was performed that involved fundamentally the SSCs which are inaccessible during the operation at full power. The evaluations of the results of that walk down are in progress.

Cordoba University is also updating the seismic data referred to the siting and the spectra in the different levels of the building. Some of the specialists of this institution took part in the seismic studies performed before the CNE commissioning at the beginning of the eighties.

It is important to point out that NASA communicated to ARN its intention of extending CNE lifetime and the safety assessment on seismic occurrence is part of the basic studies to define and authorize the lifetime extension. The studies are expected to be finished for the next Nuclear Safety Convention.

3.17.6. 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 site 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.

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 occurre.

On the one hand, these standards are compatible with deterministic concepts such as the defence in depth principle, and on the other hand, they incorporate probabilistic concepts in order to define design criteria for the plants.

The NPPs design is in accordance with the defence in depth principle and complies, in addition, with the criteria of redundancy, physical separation and diversity specified by the regulatory standards.

Besides, requirements taking into account the prevention of eventual component 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

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.

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. Such basic safety principles are included in the applicable Argentinean regulatory standards.

CNA II is at the "under-construction" stage. It is conceptually similar to CNA I, but it includes the improvements derived of the use of the Konvoi design and the operative experience gained from CNA I.

CNA II has more advanced safety aspects than CNA I, such as the redundancy of "2 of 4" in relevant safety systems, better base material of the RPV, easiness for early detection of fissures and "stellite-6" elimination.

CAREM is in the design stage, 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-pressurized primary system and safety systems relying on passive features.

3.18.3. COMPLIANCE WITH ARGENTINEAN REGULATORY STANDARDS: GENERAL ASPECTS.

Some regulatory standards were issued after the construction of CNA I and CNE, so that the Regulatory Body did not ask for their immediate application. Nevertheless, those standards are already being fulfilled or are being implemented. The Regulatory Standards are been applied to CNA II.

The fuel elements are controlled, inspected, tested and verified according to the guidelines established in each installation quality assurance program, which comprises manufacture, transportation, reception and use stages.

The primary circuit integrity for both normal and 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 are specially referred to the: number of barriers, retention capacity of radioactive material, behaviour under normal and accidental loads, and leakage rate to the atmosphere and the corresponding verification tests.

The shutdown systems (control rod insertion and liquid poison injection) 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 allows the preservation of an adequate safety level under normal and accidental conditions. It also includes the independence, redundancy, physical separation and diversity criteria. External events such as fire and missiles are also considered.

With respect to CAREM reactor, taking into account the preliminary development of this project and as it is a prototype, the ARN has started a review of an applicable regulatory frame for its commissioning, so as to analyse the design in relation to that frame.

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 3.2.1 standard. Moreover, the safety systems design complies with the single failure criterion as well as with the segregation and diversity. The latter also applies to all those systems which may require it.

The core heat removal system design complies with the requirements of AR 3.3.2 and AR 3.3.3 standards 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, in general, with the criteria established in AR 3.4.2 standard, particularly as far as diversity, redundancy and reliability concerns.

The following systems constitute CNA I confinement barriers, as required by AR 3.4.3 standard:

- The containment system*:* this system is constituted by a steel sphere of approximately 50 m in diameter wrapped up 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 accident*:* this system is located between the steel sphere and the external shield and operates by passing air through carbon and absolute filters.

CNA I design complies with the requirements of AR 3.2.1, AR 3.3.1, AR 3.4.1 and AR 3.4.3 standards, 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 components, designed and manufactured in the Federal Republic of Germany, had to comply with the requirements of German standards and that the CNA I design had to be licensable by that country.

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.

AR 3.2.1 standard criteria, related to the operator performance, are also fulfilled. The operator may always make provisions in order to avoid a situation that could affect the NPP safety, but he should not avoid the necessary operation of safety systems. In any state of the NPP, all the manually executed commands are subordinated to the reactor protection system; therefore, reactor safety is not threatened by the non-detection of measurement devise readings or alarm signals, or any eventual human error that could occur.

Taking into account the state of the art regarding information processing and report systems at the time the NPP was designed, AR 3.4.1 standard requirements concerning man-machine interface are in general 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.

3.18.3.2. CNE NUCLEAR POWER PLANT

The reactor safety system design and the contention 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 AR 3.3.2, AR 3.3.3 and A.R. 3.3.4 standards.

The core's heat removal system design comply with the requirements of AR 3.3.2 and AR 3.3.3 standards under normal operation (primary heat transport system and shutdown cooling system), and during accidents (emergency core cooling systems, high, medium and low pressure stages and emergency water supply system).

The confinement barrier required by AR 3.4.3 standard in CNE 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 AR 3.2.1, AR 3.3.1 and AR 3.4.3 standard requirements, particularly regarding the 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.

AR 3.2.1 standard 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, AR 3.4.1 standard requirements related to man-machine interface are in general fulfilled.

During 2003 the third step of the dry storage program was finalized. It included forty new silos to extend the Dry Storage of Irradiated Fuel Elements System to cover the plant's operation needs for the next four years. The capacity consists of nine canisters with 60 fuel elements per canister. The total storage capacity is 540 fuel elements.

3.18.3.3. CNA II NUCLEAR POWER PLANT

CNA II belongs to a second generation of reactors (designed in 1979) PHWR type with 745MW (e) power installed. All the systems on the nuclear area were designed in a similar way to German Konvoi PWR plants (around 1989), except the specific system of design based on heavy water (for details, see Annex VI). The operating experience of CNA I was used to improve the reactor's internals design. The Construction License was issued on 14th July 1981; and the construction process has been delayed up to now.

The Argentine Government has decided to complete the construction and to proceed to the commissioning of CNA II. This is an ongoing process under the direct responsibility of the state enterprise NASA.

Bearing this in mind, the National Executive Power Decree 981/2005 instructed NASA, as Licensee, to conform Atucha II Unit of Management, whose objective is to carry out the activities which are required to put CNA II in operation and Decree 1085/2006 keeps in use the regulations restored for the execution of the works of the NPP granted to CNA II by the Unit of Management Atucha II.

On the other hand, the commissioning activities for CNA II are carried out by ARN in the frame of the regulatory frame described in Chapter 3.7 of the present report. They consist principally in the revalidation of the Construction License, in order to update the necessary aspects for the revalidation, and performance of independent safety assessments and in the inspection of assembly and tests that will be carried our during the construction and commissioning stages.

During the period of delay, the organisation responsible for the construction has been working on several activities related to maintain the already installed equipments and those stored, as well as in the documentation related to detail engineering and the update of the Safety Analysis Report (SAR). SAR has been prepared following the US Regulatory Guide 1.70, Revision 3, November 1978, and basically fulfils the requirement on SAR´s laid down in the IAEA Safety Guide Nº 50-SG-G2. The present status of the construction reached an estimated overall completion of 81%. The schedule time for accomplishing the remaining works is estimated in about 54 months after restarting the main tasks.

The most outstanding construction activities carried out prior to government decision of speeding the works in 2006 were the following:

- *The Civil Work is practically over.*
- *The assembling of the pressure vessel, the moderator heat exchangers and devices to make the corresponding inspections to the pressure vessel surveillance system.*
- *Commissioning of the ventilation system corresponding to both the control room and the rooms containing the turbine generator and electricity distribution rod switches for its own consumption.*
- *Commissioning of the reactor Building and Turbine Building crane.*
- *Storage of components (for example main turbine) in the rooms where they will be assembled, where they were almost conditioned to normal operation.*

Before 2006, the main regulatory activities were the following:

- *Updating of the applicable ARN standards, taking into account safety evolution after 1979 when CNA II construction was decided.*
- *Evaluation of the preliminary safety analysis report.*
- *Evaluation of the preliminary risk report.*
- *Evaluation of the preliminary report of design questioning for safeguard.*
- *Verification of the operative experience transfer from CAN I to the CAN II design*
- *Regulatory inspections during the manufacturing and large components installation.*
- *Regulatory inspections carried out to the civil works and to the components storage areas.*
- *Quality Audits carried out to the Licensee.*
- *Taking advantage of the Brazilian Regulatory Body's experience in Licensing Angra II NPP, taken into account that the Designer is the same as for CNA II and that its construction was also delayed for a long period.*

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 art related to this kind of installation could affect adversely the installation's licensing process.

ARN analyzed the conditions and came to the conclusion that:

1. *There exist no restraints to continue the licensing process of CNA II, as far as the Licensee fulfils with the legal regulatory frame in force, that include specific requirements that arise from safety evaluations and inspections that will be performed in the future.*
2. *Should the Operation License be granted, the Licensee shall also fulfil with the legal regulatory frame according to Law 24.804, its Regulatory Decree 1390/98, the regulatory standards, regulatory requirements and Sanctions Regulations of ARN, and with the international legal instruments bonds 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 safeguard.*

ARN based its opinion on the following issues:

1. *CNA I is a NPP of second generation whose design dates from ´60 and has been in operation since its commissioning in 1974 with a satisfactory safety level. The installation operation for more than three decades, as well as the performance follow- up of other similar installations on the international level, allow the acquisition of great operative experience that helped, in turn, to implement significant safety back fitting improvements. Among the most relevant aspects are: a new core emergency cooling system through steam generators (second heat sink), the installation of insulating valves in the pressurized auxiliary line, the safety assessment of the RPV integrity, the new design of the moderator pool internal components and the elimination of the reactor internal's material "stellite-6".*
2. *CNA II has more advanced safety aspects than its predecessor CNA I, coming from the "Konvoi" design concept that was used. In fact, among other things it can be mentioned the redundancy of "2 of 4" relevant safety systems such as: the moderator and the essential electric supply, better base material of the pressure vessel (low copper and phosphorus content), easiness for early detection of fissures and the existence in the basic design of an emergency cooling system of the core through steam generators that function as second heat sink.*

3. *Apart from these original design safety aspects, the operative experience of CNA I and the applicable international experience should be added, such as the reactor's internals design (principally improvements in the fuel channels, in the control rod guide channels and instrumentation guide tubes), and the elimination of "stellite-6" in the core materials.*
4. *CNA II is the first NPP to whose licensing the AR 3.1.3 regulatory design standard will be applied. In CNA I and CNE II this operative was not applied because it was issued after the licensing of these installations.*

ARN personnel with the advise of domestic and foreign institutions, carry out the regulatory tasks of evaluation, inspection and audits. The counseling service are performed through contracts or specific agreements. In fact contracts have been made with institutions of USA, Germany and Canada as mentioned in Chapter 3.8.

The increase of professional personnel in ARN, as well in the hiring of third party consulting services by, implied the need of an outstanding increase in the institution's budget resources (see Chapter 3.8).

3.18.3.3.1. CNA II Preservation of Components

During the time when the construction of CNA II was delayed, one of the principal concerns was the appropriate preservation of components.

The following criteria was used for the appropriate components preservation:

- *Influence on Nuclear Safety*
- *Economical value*
- *Replacement feasibility*
- *Impact on Project schedule*
- *Preservation cost versus Replacement*
- *Damage Sensitivity*

The component preservation tasks, were divided in:

- *Routine: Applicable to all components and installations. Their scope 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 taken into account were: the materials, parts and elements affected by ageing. The most important were Gaskets, Rings, Welding Electrode, 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

Also taken into account were the materials, parts and elements affected by new regulations, such as the Insulation material for primary system components (experiences recently gained from sump clogging), change of refrigerant in chilled water machines (Ex.: R12 to R134a), and elements containing asbestos.

The preservation process was subjected to a continuous assessment by means of Licensee Internal and External Quality Audits, Siemens Inspections, Insurance Company Verifications and Regulatory Body Verification.

For example the Inspections of preservation tasks performed by Siemens were the following: June and October 1986, May 1988, February and March 1989, October 1990, February and June 1991, March and December 1992, April and November 1993, March 1995, April 1996, April 1997, April 1998, October 2003 and April 2005 (Siemens/FANP walkthrough).

Additionally, as is mentioned in subsection 3.18.3.3.2.2, 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, is one of the main concerns of NASA. In that sense the personnel that executed preservation tasks were trained and qualified according to NASA procedures, while Siemens/FANP qualified Preservation Supervisors and Preservation Team.

The component preservation process results can be summarize as follow:

- *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 have been identified*
- *Criteria of specific revision of components and evaluation of possible replacement of parts subjected to natural ageing, will be applied during the pre-phase of the Project*

The ARN is analysing all the information submitted by NASA.

3.18.3.3.2. Critical Technical Issues established by the 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 it became necessary to establish a new organization to replace it.

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, the ARN established as first priority a set of critical technical issues to be solved. The Licensee was required to provide adequate solutions on the critical technical issues mentioned below.

- *Update of the Quality System,*
- *Qualification of the Design Authority,*
- *Review of Licensing Basic Criteria,*
- *Review of Safety Issues.*

In that sense the initial action taken by ARN was to update the process required to qualify the Design Authority to be proposed by NASA in the framework of the requisites indicated in Guides IAEA NS-R-1 (design), IAEA NS-R-2 (operation) and INSAG-19.

The information given at present by the Licensee for each critical technical issue are under evaluation process by the ARN. The information under evaluation concerning each critical issue is shown in subsections 3.18.3.3.2.1 to 3.18.3.3.2.4.

3.18.3.3.2.1. Update of the Quality System

In order to comply with this requirement, in March 2007 NASA sent 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 QA original program (Overall Quality Assurance Program QAP 115 y QSP 4a and 15a/c Rev. 2 mentioned in PSAR of CNA II).

Besides its updating, the following documents were taken into account: IAEA Practice Code 50-C-Q, ISO Standard 9001:2000 and Regulatory Standard: AR 3.6.1. This upgrading comprises also occupational Health and Safety requirements that were performed taking into account 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.

3.18.3.3.2.2. Qualification of the Design Authority

With the purpose to meet this requirement and assume the project direction, during years 2004 and 2005 NASA maintained negotiations with the original designer of the plant –Siemens- in order to discuss the technical and financial conditions to resume and finalize the construction of CNA II.

According to the negotiations, NASA UG-CNA II 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 fulfill this objective.

The CNA II detail design is 90% fulfilled. The remaining 10% will be completed by NASA with the technical assistance of different institutions. Some of them are shown in the following Table.

INSTITUTION	SUBJECT
SIEMENS	*Siemens will provide supplies and services for the conventional area of the plant. Signed end of 2006*
AREVA	*AREVA will provide experts for the project in engineering, licensing, erection and commissioning areas. Signed June, 2007*
AECL	*A frame agreement was signed to provide experts, engineering packages and supplies.*
IAEA	*A technical cooperation agreement was signed between CNEA-NASA and the IAEA. The objective of the mission is an independent revision of the engineering and licensing related activities. During 2006 and the first part of 2007 two missions took place, the first mission 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. (Additionally NASA participates in a preparatory meeting to prepare a TECDOC regarding the restart of delayed NPPs projects).*
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 the construction and the licensing of CNA II. 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	*GRS shall perform a PSA Level 2 for CNA II with the necessary support of NASA. The PSA Level 2 is, together with PSA Level 1 and Level 3 (which will be both performed by NASA), a basic requirement for the licensing of CNA II to verify the compliance with the Argentinean regulatory standard AR 3.1.3. Signed beginning of 2007.*
CEN/SCK	*Integral solution for the surveillance program of the reactor pressure vessel. Signed 2007*
CNEA	*A strategic partnership between NASA and CNEA was signed. CNEA specialists are delegated to NASA to perform activities in different areas like engineering, licensing, erection and commissioning.*

3.18.3.3.2.3. Review of the licensing basic criteria

In November 1977, the Regulatory Body 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 explains that the Argentine regulatory criterion does not use the concept of maximum credible accident, the event of Loss of Coolant (LOCA) should be taken into account in the frame of a probabilistic risk analysis.

Consequently for CNA II basic design, the LOCA evaluation were carried out in the frame of the risk probabilistic analyses, out of which arose, as design base accident, a rupture in the biggest pipe connecting the primary circuit and moderator, which is smaller than rupture 2A.

From the Probabilistic Risk Analysis applied to a big LOCA (as the 2A rupture), a value less than 10-7 / year frequency of occurrence was obtained. That value was obtained by extrapolation, from the respective Biblis B NPP (German) taken as reference. The adoption of that value applied, 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 rupture 2A of the principal pipe.

The Regulatory Body objected to the bases of risk analysis, as it was not possible to make any judgment on the validity of extrapolating the occurrence frequency value to a LOCA 2A event, defined as 10^{-4}/year for Biblis B, to a lower value than 10^{-7}/year for CNA II, as the information supplied by the designer on the risk analysis was indicative rather than demonstrative. The Regulatory Body concluded that it was necessary to require more information on the LOCA 2A probabilistic evaluation.

New probabilistic risk analysis applied to big LOCA are been developed by NASA (see subsection 3.18.3.3.2.4).

Besides, it is worth while mentioning, that a review of the present predominant basic criteria all over the world resulted that LOCA 2A shall be considered as an accident that must be covered by the safety systems, independently from its occurrence probability.

3.18.3.3.2.4. Review of safety aspects

CNA II is a NPP whose original design involves that the vacuum reactivity coefficient be positive. For that reason ARN considered that it is of high importance to analyze the installation behavior in those events that may lead to steam or vacuum equivalent formation in the primary circuit, where the LOCA can be considered as most relevant.

A thorough analysis of this problem shall consider the different 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 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

This complete analysis is being carried out by ARN request and connected with this it may be pointed out:

c. *a) Compliance with the regulatory criterion defined in regulatory standard AR 3.1.3*

   - *NASA is finishing PSA level 1 which is being reviewed by ARN.*
   - *NASA is developing PSA level 2and 3 with external advising in case of level 2 and ARN is performing an on-line review of both PSA.*
   - *With the deterministic (see b2 below) and PSA results NASA expects to fulfils with what is established in regulatory Standard AR 3.1.3.*

   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.*

   Many of these evaluations are being performed with external technical advising.

   b2) Calculation performed based on an adequate and validated code

   As was mention in subsection 18.3.3.2.2, NASA has signed an agreement with the UNIPI for the development and application of a model of CNA II utilizing Relap5/3D- Nestle (thermo-hydraulic and neutronic coupling programs) and CFX (program of computer analysis of fluid-dynamic).

   Additionally, an international "ad-hoc" expert group (IRG – International Review Group) will supervise and approve all the activities developed by UNIPI for NASA. Besides it will establish a connexion between the scope of these activities and:

   - *Whether the need of experimental tests that may arise as result of this analysis are necessary or not.*
   - *The probabilistic safety analysis.*
   - *The assistance to NASA in the case of eventual requirements or request of further information by ARN.*

   On the other hand, ARN has signed an agreement with Purdue University to carry out independent assessments, based on development and application of a model of CNA II, using the programs Relap5 Mod3/Parcs (supported by US-NRC).

   b3) Improvements that can be made to the design of the prompt extinguish system

   NASA has begun the study of the System modification of a fast injection of boron and has analyzed modifications to the basic system design. NASA is now studying the material possibilities of its modifications and limitations.

   NASA is also evaluating the additional introduction of negative reactivity simultaneously to boron injection.

b4) Improvements in the design in order to get a better void reactivity coefficient

NASA is analyzing 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, NASA 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).

3.18.3.3.3. Current Design and Engineering tasks

Related to the current design the following relevant activities are being performed:

- *Design verification of the mass flow distribution in the lower plenum.*
- *Redesign of the sump filtering systems.*
- *Improvement of the Fast Boron Injection System reducing the injection delays and increasing the injection velocities.*

Regarding engineering tasks being carried out at the present time, the following jobs were initiated:

- *Completion of the detailed piping engineering.*
- *Completion of the missing supports and anchorage plates.*
- *Analysis of the consequential failures after a pipe break.*
- *Seismic analysis.*
- *Completion of the detail engineering in the civil works.*
- *Stress analysis in pipes and components missing at the time the project restart.*
- *Detail engineering for the 500kV substation.*
- *Completion of the wiring engineering.*
- *Technical specifications to perform a contract for the auxiliary steam system building.*
- *Technical specifications to perform a contract for the auxiliary reactor building.*

3.18.3.3.4. Current Construction tasks

The following table describes the current construction tasks

Heavy water	*A contract with ENSI (local company subsidiary of CNEA) was signed in June 2006 in order to produce 600 Tons of heavy water (initial amount necessary for the first load in CNA II).*
Fuel Elements	*A contract with CNEA was signed, to evaluate the design and perform permanent inspections during fabrication. The manufacturer will be CONUAR, a local company that produces the CNA I and CNE fuel elements.*
Civil works	• *Water turbine building: The excavation have been continued till level -6.00 m.* • *Cooling water intake building: The concrete of beams has been continued.* • *Service water system for secured plant: Excavation and transport of soil.* • *Cooling water seal pit and service cooling water collecting pit: The excavation for the wall-plate and anchorage plates has been continued.* • *Construction details in the following buildings: Containment Inner Structure and Reactor Auxiliary Building, Fuel Storage Pool Building, Staff Facilities and Offices Building, and Demineralizing System Building.*
Electromechanical erection	• *Reactor Auxiliary Building:* o *Prefabrication and assembly of pipes* o *Prefabrication and assembly of Frames of I&C.* • *Inner Containment Structure* o *Posts for cables and solid bottom trays for cables of measurement of neutron flux were erected.* o *The alignment and leveling of the Main Coolant Pumps housing was finished.* o *Analyses for the erection of the Primary, Moderator and Main Steam systems.*

	• *Reactor Building Annulus*
	  o *Monorails and eyebolts welding.*
	  o *Positioning of frames of I&C.*
	• *Reactor Room*
	  o *Training of welders for the Primary System was performed.*
	  o *Analysis of existing prefabricated pipes/supports of the systems in the building. Comparison with drawings*
	  o *It is foreseen to begin the erection of the Primary and Moderator Systems in September 2007*
	• *Turbine Building -*
	  o *Analysis of existing prefabricated pipes/supports Comparison with drawings*
	  o *Pipes of DN 2200 were analyzed, previous to the assembly of the condenser*

Components revision	*The following components are under revision by manufacturer personnel:* • *Main components of the water (Siemens).* • *Main Coolant Pumps parts (Andritz)* • *Valves parts (Sulzer)* • *Primary system tools*

3.18.3.4. CAREM REACTOR PROTOTYPE

The CAREM concept that belongs to the very low or low power nuclear plants[1], was put forward from the very beginning as an advanced design reactor, being the precursor of innovative concepts as regards safety. It is a light water reactor that uses as fuel enriched uranium with new design solutions based in the wide experience accumulated worldwide regarding the safe operation of light water reactors; it has direct cycle particularly simple in its conception, which contributes to its high level of safety, being the followings its main innovative aspects:

- *Integrated Primary System*
- *Self pressurization*
- *Safety Passive System*

Even though both technical and engineering solutions associated to the NPPs technology and the innovative design characteristics are correctly verified during the design phase, since CAREM concept has innovative characteristics it considered convenient to construct a reactor prototype to validate its design, manufacturing, installation and operational aspects as well as verification of component and systems reliability. A more detailed description will be found in Annex VII.

Due to this fact, the CNEA as the head organism of the nuclear activity of Argentine Republic has proposed the National Government to carry out the construction of the CAREM Reactor prototype, by means of the construction of the CAREM Nuclear Plant of 25MWe power. The size of the before mentioned reactor has been selected taking into consideration the following reasons:

- *It is the minimum electric power 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 inversion required depending on the very low power that it has.*
- *It is a reactor that has good possibilities of being commercialized without the need of modifications for the introduction of nuclear energy in developing countries, since it has costs comparable to the ones of the research reactors.*
- *It has a size that allows a relatively easy change of scale to powers with which it will be possible to supply isolated areas and to satisfy the requirements of the distributed generation (in the range from 25 to 30 MWe)*

The CAREM project was initiated in Argentina more than twenty years ago, and the original objective was to study the possibility of filling a gap generated in the nuclear industry: the one of the very small and small reactors to facilitate the income of electric generation of nuclear origin for countries that need to give their first steps in this field and with which Argentina has a long collaboration trajectory.(Although

[1] IAEA defines the reactors, depending on the power: Very Low Power <150MWe, Low Power 150-130MWe

the power might be good choice for countries that are needing to increase a small fraction of their current electrical production).

This project has enabled Argentina to make an incursion into the area of NPPs 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 held in March 1984. Since then, some of the design criteria of CAREM have been used by other designers, originating a new generation of reactors, which the CAREM is chronologically one of the first and one of 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 nucleoelectric generation[2], including the CAREM. Argentina is among those 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 (USA); EPRI (Electric Power Research Institute, USA); INEEL (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[3]:

- *It has an evolutional design available for its construction in the short term.*
- *As regard safety it is outstanding.*
- *From the economic point of view it is above the average.*
- *The fuel utilization and the mechanisms handling are advanced ones.*
- *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[4].

3.18.3.4.1. Construction of CAREM Prototype

The "Secretaría de Energía" of "Ministerio de Planificación Federal, Inversión Pública y Servicios" (Secretary of Energy, Argentina) has promoted the Decree of the Argentine President Nº 1107/2006. By means of that decree, CNEA and EBISA (Emprendimientos Energéticos Binacionales S.A, Argentina) have been instructed to follow the necessary actions for the prototype construction. EBISA will contribute to the project with its experience in managing great binational projects.

CNEA, as the owner of the technology of CAREM Plant, will look after safety matters as well as planning and construction solutions. CNEA as a Technical Support Organization will provide scientific and technological backup to the works related to the project and will be responsible for the proper implementation of tests and qualification of fuel elements and safety components of the CAREM.

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2 U.S.S DOE Nuclear Energy Research Advisory Committee and the Generation IV International Forum: "A Technology Roadmap for Generation IV Nuclear Energy Systems" GIF-002-00, December 2002

U.S. DOE Nuclear Energy Research Advisory Committee and the Generation IV International Forum: "R&D Scope Report for Water-Cooled Reactor Systems", GIF -003-00, December 2002

3 U.S. DOE Nuclear Energy Research Advisory Committee and the Generation IV International Forum: "Description of Candidate Water-Cooled Reactor Systems Report", GIF-015-00, December 2002.

4 [5]OECD/IEA: "INNOVATIVE NUCLEAR REACTOR DEVELOPMENT, Opportunities for International Co-operation" NEA 3969 Paris, France, 2002

CNEA will be responsible for the Commissioning of the CAREM.

According to the CNEA Management Systems engineering changes will be approved by CNEA.

In line with the integration of national capacities, the participation of domestic companies that have a technological capacity for this kind of undertakings, and contribute to the project investment is being analyzed. Up to now, the following companies were selected INVAP S.E and IMPSA (Industrias Metalúrgicas Pescarmona S.A.I.C. & F., Argentina). Both can play an important role in the fulfillment of the CAREM-25 Prototype construction and commercialization. Both can complement each other to cover engineering and manufacturing of heavy components.

CNEA has generated another series of companies with great technical capacity in the nuclear area that will play an important role in the CAREM-25 Project, they are: CONUAR, FAE, DIOXITEK, ENSI and NASA.

Nowadays CNEA and EBISA are carrying out actions for the creation of a new Company CAREM S.A. This company will be responsible of the administration and management of the construction of CAREM 25 prototype. The companies mentioned above will participate with different shares in CAREM S. A. participating in the project to contribute with budget and with their engineering and construction capacities. This spectrum of domestic companies will be enlarged with others that can contribute with their economic and technological capacity to the different areas of the project, always complying with the parameters of quality and nuclear safety.

This prototype should be in operation by 2012.

3.18.3.4.2. Budget and Human Resources

Recently the Government has assigned budget to CNEA to begin the tasks for the construction of the Prototype CAREM 25 during 2007, funds were also assigned for the 2008, 2009 and 2010 budgets.

During the current year, tasks of Project Organization, creation of a sector of Quality Assurance for the Project, and upgrade of the engineering documentation has begun. In addition computer tools were incorporated for design and to facilitate the handling of the documentation. Activities on a test loop for the drive mechanisms and other safety components in continuing with IAEA support.

The Government authorized for the 2007 Budget to incorporate professionals from the CNEA´s Institutes, such as Nuclear and Mechanical Engineers from Instituto Balseiro, Specialist on material science from Instituto Sábato and reactor fuels from Instituto Dan Beninson. A workshop has also been carried out on Project Management in the Institute Balseiro with support of the IAEA. Professional youths that are working are scheduled to participate in specific modules of the Nuclear Safety international course organized by the ARN.

3.18.3.4.3. Siting and Licenses

The Honorable Camera of Deputies of the Buenos Aires Province has manifested its interest to have the CAREM-25 prototype constructed in Lima, Province of Buenos Aires, in the lands of CNEA, next to CNA I and CNA II. As a prototype, a specific license process will be defined. To make an initial evaluation, the ARN requested a draft Preliminary Safety Report which has already been sent. It is expected that after the preliminary assessment by the ARN (to evaluate among other actions, the regulatory effort that will be needed), the CNEA will be in condition to request the Construction Permit, estimated to be issued by the end of 2008.

Taken into account the preliminary state of development of the CAREM´s proyect and due to its prototype character, the ARN has started, as a first step, a detailed revision of the regulatory framework applicable for its licensing.

3.18.3.5. FOURTH NUCLEAR POWER PLANT

The fourth NPP is at its pre-feasible stage of study. At present the siting assessment is being performed as well as the environmental impact. It concerns one (or two) NPP CANDU 6 E types, each one with a net power of 750 MW(e).

3.18.4. COMPLIANCE WITH THE OBLIGATIONS IMPOSED BY THE CONVENTION

In Argentine, 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 occurre.

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 authorised 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 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 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, In-Service Inspections, programmed maintenance and any other safety related activity.

The Reactor Manager is supported by an engineering section giving part of the technical support needed for the NPP operation. Besides, the Licensee also has an engineering division satisfying some of the technical support 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 AR 3.9.2 standard as well as with other applicable regulatory requirement.

The feedback process of operating experience of domestic NPPs involves the following entities: Licensee, ARN, Designers, Component Suppliers and international organisations dedicated to information distribution.

Finally, could be mentioned that 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 AUTHORISATION TO OPERATE

3.19.2.1. CNA I - INITIAL AUTHORISATION TO OPERATE

In May 31st 1968 a contract between CNEA and Siemens was signed for the construction of CNA I. It was established that concerning radiological and nuclear safety, the design should comply with standards, rules and laws in-force of 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 Organisation Technischer Üeberwachungs Verein, Baden (TÜV).

In 1971, the TÜV Baden issued a report concerning CNA I construction, mainly containing a series of requirements, recommendations and additional information requests. It also carried out inspections to the fabrication 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 held the responsibility of carrying out the test 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 additional information requests still pending, as well as advising their authorities in what concerns CNA I and its operation personnel licensing process.

3.19.2.2. CNE - INITIAL AUTHORISATION TO OPERATE

CNE initial authorisation was issued according to the requirements established in AR 3.8.1 and AR 3.8.2 standards. The first is related to pre-nuclear commissioning and establishes that the Licensee must have a Pre Commissioning Program and an Organisation to carry it out. The pre-nuclear commissioning program comprises those tests required to demonstrate the safe operation of the NPP.

AR 3.8.2 standard also establishes that the Licensee must have a nuclear commissioning program and an organisation 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 design, construction and operation of NPPs. The ad-hoc committee has the main responsibility of evaluating each of the stages the commissioning program is divided into, and authorises the transition from one stage to the other.

During pre-nuclear and nuclear commissioning stages, the ARN verified that the Licensee complied with the mentioned standards.

3.19.2.3. CNA II - INITIAL AUTHORISATION TO OPERATE

In Chapter 3.18 of the present report, those aspects related to CNA II commissioning are developed.

3.19.3. OPERATIONAL LIMITS AND CONDITIONS, MAINTENANCE, AND TESTING

3.19.3.1. CNA I - OPERATIONAL LIMITS AND CONDITIONS, MAINTENANCE, AND TESTING

The conditions for the authorisation 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, authorised 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 spread out in different documents, such as the SAR, the Operating Manual, and the Maintenance Manual- has been collected in the Policies and Principles Manual. The Policies and Principles Manual of CNA I establishes the ranges of valid values some plant operational parameters must comply with other specifications, as well as the organisation 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 AR 3.9.1 Standard), the Safety Advisory Internal Committee activities and the communication of significant events to the ARN.

CNA I has preventive maintenance and in-service inspection 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, which are part of the mandatory documentation required in the Operating License.

The surveillance program including in-service inspection activities related to significant components, equipment and systems are routinely carried out, mainly involving the following: pressure vessel; primary, moderator and volume regulation systems, as well as steam generators and moderator heat exchanger tubes.

During the informed period CNA I Operational Limits and Conditions were modified regarding upper allowed values for: a) turbine vibration; b) fuel burn up, and c) neutron flux measured by in core detectors. These changes were included into the corresponding operating procedures.

Additionally, a new version of CNA I In Service Inspection Manual was performed considering categorization of inspected elements into two categories: "obligatory" and "conditioned" and following the backfitting process carried out up to 2004, 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 authorisation of commercial operation of CNE have been mainly established in the Operating License. In fact, in the license, the essential requirements for the installation operation such as maximum reactor thermal power, limits of authorised discharges, communication of the occurred significant events to the ARN, etc. are explicitly contained or refer to other related documents.

Another conditioning requirement for CNE commercial operation was the Regulatory requirement CALIN 122/84 document (see first Argentinean Report to the Convention on Nuclear Safety-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 operational procedures.

CNE has preventive maintenance and in-service inspection 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, which are part of the mandatory documentation required in the Operating License.

The surveillance program including in-service inspection 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 and CNE Inspection Programmes were described in detail in the previous National Nuclear Safety Report. In the following paragraphs, the most important inspection activities performed during the last period in both NPP are detailed.

3.19.4.1. CNA I - INSPECTIONS

3.19.4.1.1. In Service Inspections

The In-Service Inspections in CNA I were performed during this last period in agreement to a previously adopted schedule. Looking for a more orderly report of results, the most remarkable data obtained from inspections and those concerning Reactor Pressure Vessel are detailed in Chapter 3.14.

3.19.4.1.2. RPV Internal Components Surveillance Program

Once CNA I backffitting was completed, there was a regulatory requirement for the development of an RPV internal components surveillance program.

According to this program, twelve positions were selected to take out the corresponding channels for the inspections. 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 can be added as a result of the "early failure detection program".

The surveillance program started in the 2006 planned outage. The results from the inspections were satisfactory. One channel was taken out for destructive tests that are being carried out by CNEA.

3.19.4.1.3. Heat Exchanger RR42W02 (Safety Heat Removal System)

The heat exchanger RR42W02 is part of the safety heat removal system (RR), which includes both moderator circuits, each one cooled by one of the RR system circuits. Residual heat is then removed from the RR system by the safety river water cooling system UK. The heat exchanger RR42W02 is the interface between RR and UK systems in one of the two circuits. The safety heat removal system is used to cool down the plant till Cold Shutdown State.

Besides, the RR system is connected during normal operation to the steam generator feedwater system (RL). Recently, a leakage from the RR-RL circuit towards the UK system was detected obliging to repair the heat exchanger. The repair was performed, although partially, during May 2007. The leakage was due to failures in heat exchanger tubes due to high stress in the rolled joint zone, and cracks in the tube sheet. In the next planned outage (2008) the heat exchanger tube bundle and tube sheet will be changed, giving a definite solution to the problem.

3.19.4.1.4. 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 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 inspectors and analysts of the ARN.

A Fire fighting system inspection was carried out by the Regulatory Body during the outage of CNA I in 2006. The job aimed at design aspects, along with the protection and maintenance measures, for the sake of verifying the safety level of the plant in the fire fighting issue.

In order for the inspection to be accomplished, the safety radiological team of the Federal Bureau firefighting body was required to take part, along with the staff assigned by the ARN.

The inspection consisted in the verification of the application of the AR 3.2.3 standard, the IAEA Safety Guide Nro. NS-G-2.1 for the implementation and development of an adequate protection program against fire, the CNA I PSA Phase 2 Study for Internal Fires that identified fire zones, its frequencies and propagation criteria, as well as the PS 63 procedure "Conventional Emergencies", which is in force at the plant, and the respective periodic control protocols.

The most important inspection recommendations and the corresponding implementation are:

- *To increase training frequency, from yearly to no more than 6 month periods and fire drills and person rescue theoretic-practical courses should be included. Re-training routine in these issues are supplied by a fire-fighting highly qualified external team.*
- *Fire-fighting availability: to incorporate a team of a minimum six persons, available 24 hours per day, since at present there is a standard unit and a two firemen team. The plant has started conversations with the firemen body of the Federal Bureau, in order to reach an agreement on issues related with the human resource supply.*

- *To perform an exhaustive inspection of the system in order to obtain a certificate to guarantee adequate function in emergency situations. The plant has begun the actions to update and centralize the systems, jobs that will be done by an external contractor.*
- *To analyse the possibility of anti fire painting/liner in the main control room in order to diminish fire propagation possibility . The plant has started the analysis.*
- *To construct a container capable of retaining 110% of the fuel in case there is a spread in the diesel generator room. This recommendation is still under analysis because of the fact that daily fuel tanks have a double wall.*

Regulatory inspections covering Embalse and Atucha II fire protection systems will take place in the near future.

3.19.4.2. CNE - INSPECTIONS

3.19.4.2.1. In Service Inspections

The In-Service Inspections in CNE were 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 and feeders are detailed in Chapter 3.14.

3.19.4.2.2. Steam Generators

The CNE steam generators (SG) 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 a degradation mechanism known as Flow Accelerated Corrosion (FAC), which was not well known al the moment of design and construction, and after more than 20 years of operation, the supports started to degrade producing steam generator 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 steam generators as part of the Life Management Programme. In so doing, NASA contracted B&W, who designed auxiliary supports known as antivibration bars (AVB's,). These components consist of three comb-shaped supports, for each steam generator, 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.

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

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

Taking into account the regulatory requirements and findings during visual inspections in secondary side B&W was required to implement, during 2005 planned outage, four inspection ports between the upper TSP in each steam generator, in order to inspect and evaluate the TSP state, and eventually, decide their cleaning.

From these new ports, inspections were performed that showed sludge obstructions in hot leg side of TSP, and variable degradation in cold leg side of TSP (the higher TSP was, the more important degradation was observed). 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 cold leg side of TSP, the ARN required NASA an assessment on structural integrity in order to verify the steam generators safe operation. The assessment was given to AECL, and included evaluations performed specifically for CNE:

- *Cathena Analysis for a Steam Main break at a Remote Location,*
- *Fretting and Fatigue Analysis,*
- *Fatigue Analysis Under Blow Down Conditions,*
- *Condition Assessment of CNE SG4 Degraded TSP,*
- *Thermal hydraulic Analysis with Degraded and Fouled TSP,*
- *Root Cause Analysis of the TSP FAC Degradation;*
- *Stress Analysis of Degraded Tube Support Plates Under Seismic Plus Remote Burst Pipe Loading,*

From July 2006 to February 2007, there were three unplanned outages due to tube failures in all cases localized in the U-bend zone. So the plugging criterion was changed to a more conservative position, and to extend the inspection zone taking into account the observed defects, going further than what was required by regulations.

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

- *Waterlancing of 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 pH. Besides a Cooper monitoring program was implemented, as well as a more exhaustive cation monitoring in the steam generators blow down.*
- *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 the preliminary results obtained in visual inspections that were being performed, and considering unplanned outages due to tube failures, the ARN decided to re-assess operative conditions in steam generators, to perform a detailed analysis of degradation rate, to assess the structural integrity considering the steam generators current state and including the consequences of seismic occurrences and other events that cause sudden depressurisation of secondary circuit.

Consequently, the main tasks to be undertaken by NASA in short and medium term, related to steam generators safe and reliable operation, consist of additional assessments concerning structural integrity, taking into account the new degradation maps obtained in 2007, evaluating the current condition of TSP and comparing it with the result of previous inspections. This is 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 will be considered.

3.19.4.2.3. Turbine

In 2004 a general inspection of the CNE turbine took place and from the data gathered during that and previous inspections, some of the blades from the last row were changed.

The inspection was ocular, and non-destructive tests of the components, including the blades and rotor were undertaken and ultrasonic inspection was made to verify the coupling between the blades and the disks on the ten most critical disks.

The ultrasonic inspection made with the purpose of verifying the state of the coupling between disks to the axles, showed an increase in a fissure of axial orientation at the base of disk #5 on the turbine side that had been detected during a planed outage in 2000.

Taking into consideration safety issues in the case of an eventual break of the disk, ARN required NASA an evaluation to determine the state of the turbine, detailing the criteria and the basis of acceptance used that would allow the safe operation until it should be inspected again.

The evaluation by fracture-mechanics was made by TECNATOM S.A. and revised by ANSALDO ENERGIA (the turbine manufacturer) and audited independently by ARN. The conclusion was reached *that the maximum recommended time it could operate was 15 months (taking into consideration the depth of the defect reported in the inspection, critical size and verified speed of growth).*

Finally, within the stipulated time, the disk was changed and a new one placed.

3.19.5. OPERATIONAL PROCEDURES IN NORMAL AND ACCIDENTAL CONDITIONS

Most of the CNA I operational procedures, either in normal or accidental conditions, are included in the Operating Manual. These 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 operational state, and instructions to perform infrequent hand made actions.*
- *The third part includes the manual of warnings and alarms of all the installation 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 and 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 Severe Accident Management Program are included in subsection 3.14.3.1.2

3.19.6. ENGINEERING AND TECHNICAL SUPPORT

CNA I and CNE Plant Managers have their own engineering sections. These sections are complemented with the NASA headquarter's technical services, which include specific subjects such as instrumentation-control and civil engineering, having 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 asks the CNEA for service and specialised advice (technical support). Frequently they have also used the services of INVAP S.E. (an Argentine technology organisation dedicated to high technology projects).

They have also used and will keep on using, if needed, the advice of foreign organisations, like 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.

3.19.7. INCIDENTS REPORTING

One of the main concerns of the ARN is the occurrence of significant events and the actions related with it, considered part of the profit from operating experience in NPPs.

To this respect, AR 3.9.2 standard sets the basic criteria concerning definitions, event communication modes to the ARN, and event analysis. Such analysis includes determination of root causes and effective remedial and corrective actions commensurate with the situations.

On the other hand, the Operating License sets particular conditions referred to the subject and some specific requirements have been issued concerning it.

During the 2004-2006 period, 19 relevant events were reported by CNA I and CNE. A total of 15 root / contributing causes were identified. From those, 7 events with root contributing causes related with human factors have been found. Some of them were the main cause of the occurrence of the event.

The most significant operational events in CNA I and CNE and how the Licensee and the ARN acted are given in Annex VIII.

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, transferred 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.

3.19.8.1. FEEDBACK FROM LOCAL OPERATING EXPERIENCE

CNA I and CNE have, as part of their internal organisation, 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 standard AR 3.9.2,
- Unforeseen outages,
- Minor events (CNA I) or reportable events (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 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.

In CNE, the reportable events are analysed according to a procedure, by the section in which it was originated. The recommendations or corrective actions, their implementation and distribution are a responsibility of the section involved.

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 distribution.

All the operational incidents, significant and minor events, their corrective actions and their follow up are recorded.

Every event implying an unforeseen outage and/or a deviation of operational limits and conditions established, must in addition be evaluated by the "Internal Safety Advisory Committee (CIAS) of the plant, according to what is established in standard AR 3.9.1. Its conclusions and recommendations are written down in minutes signed by the participants.

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 participants.

The significant events are communicated to the ARN according to standard AR 3.9.1 and afterwards an analytical report is issued according to dates and style established in the plant's Operating License.

When applicable, the ARN following a procedure, promptly notifies the international community of the occurrence of a significant event together with its category according to the IAEA 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.

A list of the events, lessons learned and corrective actions resulting from national and international operating experience is detailed in Annex VIII.

Finally, and as an example of the operating experience feedback of the nuclear power plants in operation in the country, the transfer to CNA II of those problems detected during CNA I's operation should be mentioned.

3.19.8.2. FEEDBACK OF THE OPERATING EXPERIENCE OF 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-II, Wolsung-II, 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 International Reporting System.

The processing of information provided by the different sources is heterogeneous and not always profitable, as it essentially dependents of the characteristics of the plant's design.

CNE uses COG databases as part of its usual working activities. The Engineering Section is in charge of carrying out a first selection of the information and transmitting it to the sections involved. 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.

CNA I has been using the WANO database since 1996. The collection, selection and classification of information have been systematised.

During the ’90, the ARN examined the effectiveness of operating experience feedback using information coming from national and international databases. As a result of this review it was decided to enhance the evaluation of incidents coming from domestic and foreign plants.

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 are 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 case that 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 optimise 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 NPP obtained from the feedback of operating experience.

A set of activities to be carried out to fulfil the program was defined and trend analysis, workshops to share experience and training were also included. The Licensee prepares a quarterly report including the results obtained by the application of the program. Besides, NPP’s senior teams evaluate “low level events” and “near misses” obtaining 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 ARN for evaluation. The evaluation could include recommendations, proposed design modifications, changes in procedures and training courses for operators if necessary.

The program coordinator reviews more than 500 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.

During the 2004-2006 period, the training courses were increased, using the lessons learned from events, organizing training meetings and discussions.

In 2006 the Regulatory Body performed an audit to the CNA I Operating Experience sector. The result show that during 2005 there had been an improvement in the follow up of corrective actions , implementation, organization of training meetings and discussions as well as an increasing experienced operating personnel.

There are many recovery actions coming from the feedback of the National Operating experience and Operating experience from Foreign NPP 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 .

Examples of corrective actions coming from local and international operating experience in the last years are shown in Annex VIII.

3.19.8.3. PEER REVIEWS AND ACTIVITIES BETWEEN THE LICENSEE AND WANO FROM 2004 TO 2007

The Licensee is a member of the WANO created in 1988, at the beginning through the CNEA and then ratified, in Paris 1995, its condition of associate as NASA.

Both NPPs under operation received WANO Peer Review Missions (CNA I NPP in 2006 and CNE at present, September 2007). NASA participated in WANO programs through the WANO – PC (Paris Centre): Peer Review Missions, Technical Exchange Visits, Operative Experience Exchange Program, Lectures and Participation in Courses and Symposiums. Argentina has also provided specialists from both NPPs to participate in Peer Review Missions.

Main activities are listed below

PERIOD	***HOSTESS***	***PARTICIPATION POSITION/PEER AREA***	***ACTIVITY***
June 14-July 2, 2004	*Wylfa NPP United Kingdom*	*Chemistry-Maintenance.*	*Peer Review*
September 6-24, 2004	*Hunterston B NPP United Kingdom*	*Engineering*	*Peer Review*
October 2-24, 2004	*Emsland NPP-Germany*	*Chemistry*	*Peer Review*
Nov. 30-Dec.4, 2004	*Naantali, Finland*	*General Manager*	*WANO Senior Executive Seminar 9*
February 11-15, 2005	***Atucha I NPP and Embalse NPP***		***WANO Technical Support Mission (TSM) TSM–05–026–AM-RR***
June 2005	***Atucha I NPP and Embalse NPP***		***Final WANO PC Report***
October 9, 2005	*Budapest*	*Governor*	*35th WANO PC Governing Board Meeting*
December 12 -15, 2005	***Atucha I NPP***		***WANO PC Peer Review Pre visit***
December 12 -15, 2005	*OSKARSHAMN*	*Operating experience*	*Peer Review*
March 2-3, 2006	*WANO– Paris Centre*	*WIO*	*34th WIO's Meeting*
March 6-24, 2006	*DOEL NPP (Belgium)*	*Maintenance*	*Peer Review*
March 22-23, 2006.	*WANO– Paris Centre*		*Root Cause Analysis Seminar, WANO PC*
March 27-29, 2006	*Rio de Janeiro, Brazil*	*Governor*	*WANO PC 36th Governing Board Meeting and 19th General Assembly*
June 26 - July 14, 2006.	***Atucha I NPP***		*Peer Review*

PERIOD	HOSTESS	PARTICIPATION POSITION/PEER AREA	ACTIVITY
September 11-29, 2006	*Sellafield (UK)*	*Radiological Protection*	*Peer Review*
September 14-15, 2006	*WANO – Paris Centre*	*WIO*	*35th WIO's Meeting*
Sep 20-22, 2006	*Switzerland*		*3rd Communication Expert Group Meeting*
October 5- 6 2006	*Neuilly sur Seine, France*	*Governor*	*37th WANO PC Governing Board Meeting*
October 11-13, 2006	*WANO – Paris Centre*		*Performance Indicators Workshop*
October 22-25, 2006	*Spain*		*6th Training Managers Expert Group Meeting*
October 27, 2006	***WANO – Paris Centre***		***Exit Meeting Atucha I NPP***
November 13-15, 2006	*London*		*Plant Managers' Conference*
Jan. 14 – Feb.1, 2007	*Madras India,*	*Engineering Support*	*Peer Review*
Feb 26 – March 2, 2007	***Embalse NPP***		***WANO TSM on Peer Review Methodology***
March 8 - 9, 2007	*WANO Paris Centre*	*WIO*	*36th WIO's Meeting*
March 22 & 23 , 2007	*Madrid*	*Governor*	*38th WANO PC Governing Board & General Assembly*
March 27 , 2007	*Spain*		*6th Human Performance Expert Meeting*
May 23-27, 2007	***Embalse NPP***		***Peer Review Pre-visit***
May 21 –25, 2007	*Almaraz NPP – Spain*		*Technical Visit*
June 11- 15, 2007	*Laguna Verde NPP Mexico*		*Technical Visit*
June 19- 20, 2007;	*Embalse NPP*		*WANO TSM on Safety Culture*
June 21 -22, 2007	*Atucha I NPP*		*WANO TSM on Safety Culture*
June 21- 22, 2007	*Paris, France*	*Atucha I NPP Manager*	*Site Managers' Seminar*
June 25 - 29, 2007	*Atucha I NPP*		*WANO AV on Operating experience Management*

In order to improve some particular areas in CNA I, after the 2007 Peer Review, NASA requested WANO assistance through several technical support missions.

- *Safety Culture (done),*
- *Operating experience (done),*
- *Industrial Safety,*
- *Task Observations (actions in the field),*
- *Radiation Protection.*

As well as follow up and support periodical visits every four months in the plant.

Besides, WANO Peer Review follow up mission to CNA I was requested for March 2009.

A CNE Peer Review is expected to take place in September 2007.

A WANO Technical Meeting on Self Assessment for CNA I and CNE personnel is planned for the near future.

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 Nº 24804, that regulates the *nuclear activity and other activities, and Law Nº 25018, that lays down the Radioactive Waste Management Regime.*

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 following paragraphs, excerpted from the National Report to the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management (2005) present the Radioactive waste management policy:

- *"The determination to manage radioactive waste originating from domestic nuclear energy applications, including wastes from the decommissioning of related facilities.*
- *The allocation of responsibilities for the performance of waste management activities, and essentially the long term surveillance and institutional control required by different final disposal systems used.*
- *The assurance that such management activities are and will be performed safely, ensuring the protection and the rights of present and future generations and the environment.*
- *The development of a Strategic Plan which is periodically reviewed, authorized and audited by the National Congress.*
- *The establishment of a proper procedure to obtain and manage the necessary financial resources to comply with the obligations arising from the performance of the assigned responsibilities, considering that many of them imply deferred costs.*
- *Maintenance of a recording and information system which provides a total knowledge and control of inventories of radioactive waste generated, and to be generated,from all nuclear activities.*
- *Development of a public communication and information program."*

3.19.9.2. SPENT FUEL MANAGEMENT POLICY

The following paragraphs, excerpted from the National Report to the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management (2005) present the Spent fuel management policy:

- *"The Argentine Government exercises the state ownership of special radioactive fission material contained in SFs originating from the operation of NPPs and from experimental, research and/or production reactors. (Act 24804, Art 2).*
- *The decision to reuse fissile material contained in SFs or not, has to be adopted before 2030. At such time it will be essential to have the required technical and human resources to implement whatever decision is adopted. (Strategic Plan – Act 25018)."*

3.19.9.3. RADIOACTIVE WASTE PRACTICES AT NPPS

Radioactive waste management at NPPs was described in detail in the First and Second National Report to the Joint Convention on the Safety of spent fuel management and on the safety of radioactive waste management (2003 and 2005). A brief summary follows.

Radioactive liquid waste are transitorily stored at each NPP awaiting treatment on site. The storage period depends on the activity and treatment technology.

Considering the different technologies used at each NPP for the treatment of low level radioactive liquid waste, herein below follows a brief description of such technologies:

- *Liquid radioactive wastes originated at CNA I during operation and maintenance activities are collected in tanks, and measured to decide if they are dischargeable (according to pre-established procedures and within authorized constraints of discharge) or needs further treatment by evaporation. Concentrates and sludge from the cleanup of tanks are immobilized in cement matrixes for their conditioning in 200 liter drums.*
- *Liquid radioactive waste originated at CNE during operation and maintenance activities are retained by filters and ionic exchange resin beds, discharging into the environment the exhausted*

stream, based on pre-established procedures after measurement, and within the authorized constrains of discharge.

Solid low level radioactive waste at both NPP´s are classified as compactable, non-compactable and structural. Compactable solid waste are collected in plastic bags and further compacted in 200 L drums. Non-compactable and structural waste are disassembled and sectioned prior to the conditioning in containers, if it is considered necessary, on a non–routinely 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. Such intermediate level radioactive solid wastes are stored at the facilities of each NPP.

All the storage facilities are placed at the NPP sites. Radioactive waste will be stored in such temporary facilities until disposal facilities be in operation.

3.19.9.4. MINIMIZATION OF RADIOACTIVE WASTES

The policy of the NPP´s Licensee is to minimize the impact on the environment as a result of its operation. Therefore, one of its main goals is to keep to the minimum practicable the generation of radioactive waste, 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 in high radiation fields and waste generation.*
- *Compliance with segregation procedures of contaminated wastes.*
- *Damaged fuel elements are immediately withdrawn from the core*
- *Operation power is usually maintained in a stable regime, with power ramps when needed, according to design.*
- *Personnel training in the application of radioactive waste management procedures.*
- *Measurement, classification, 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 amount of ^{60}*Co activity in operational wastes.*

Further details are presented in the First and Second National Reports to the Joint Convention on the Safety of spent fuel management and on the safety of radioactive waste management (2003 and 2005). Any further information can be obtained from the above mentioned National Reports that can be downloaded from:

http://www.cnea.gov.ar/xxi/residuos/residuos.asp *for Spanish version and*

http://www.cnea.gov.ar/xxi/residuos/convencion-conjunta.asp *for English version.*

3.19.10. COMPLIANCE WITH THE OBLIGATIONS IMPOSED BY THE CONVENTION

The information contained in this and other Articles demonstrates that the Argentine complies with the obligations imposed in Article 19 of the Convention on Nuclear Safety.

CHAPTER 4

PLANNED ACTIVITIES TO IMPROVE SAFETY

The ARN and NASA identified a number of challenges during this report period. Those challenges lead to actions both with regard to safety-related issues and regulatory issues in order to maintain and improve the safety level of the Argentinean nuclear power plants during their remaining operating lives, as well as to improve the regulatory framework. In the following, the intended measures are presented in a summary form.

4.1. HARMONIZATION OF ARN STANDARDS WITH IAEA SAFETY DOCUMENTS

During 2006, ARN decided to initiate an harmonization process of their standards with the IAEA safety documents. The detailed comparative analysis is well advanced, and comprises the criteria established in the ARN standards and the corresponding indicated in the IAEA Safety Requirements. The comparative analysis is also extended to the recommendations that are contained in the Safety Guides, which may be important to incorporate in ARN standards.

Some improvements opportunities have been identified in no essential aspects of the ARN standards, which will be incorporated following the corresponding procedure.

For the next Convention it is foreseen that this task will be finalized.

4.2. DEVELOPMENT OF NEW STANDARDS

In order to carry out the process of CNA II's licensing, the regulatory activities related with CNE's life extension, the future licensing of CAREM and the fourth NPP, the ARN has considered it opportune to intensify the revision of its regulatory standards and guides related to NPPs, as well as the elaboration of the necessary new standards.

In this context, it is under analysis the elaboration of new standard for NPPs' life extension and the licensing of new installation as the CAREM prototype.

4.3. QUALITY MANAGEMENT SYSTEM IN THE REGULATORY BODY

The QMS in the ARN is being implemented by applying a processes approach. Seven regulatory processes (nuclear installations, radioactive installations, transport, radioactive waste management, prevention of intentional events, special projects and non-proliferation) and four support processes (institutional communication, human resources, R&D and financial and material resources) have been identified.

The sequence and interaction of these processes are established and represented in charts. Each process is described in a Process Letter where the objectives, inputs, outputs, checkpoints, associated documents, performance indicators, non-conformances and corrective actions are considered.

The analysis and measurements of these processes are carried out by self-assessments, and Effectiveness and Efficiency Indicators are been defined.

Many of these activities require significant effort and are novel in application in Argentina. It is likely that these activities will be ongoing for some years before any safety benefits will be achieved.

4.4. TRAINING OF HUMAN RESOURCES IN THE REGULATORY BODY

ARN is presently going through a period of a marked reduction of specialised personnel due to the fact that some are reaching retirement age and to the increasing offer of better-paid job opportunities from other industries.

Furthermore, the increase in the licensing tasks for CNA II, CNE life extension and CAREM licensing as well as the pre-feasibility studies for the Forth NPP produce the big challenge of incorporate new personnel and consequently the need to train them.

In this context, the ARN is organising supportive courses for the incoming personnel and special courses for managers and is using knowledge management as an useful tool to transfer knowledge, as well as contracts and work agreements with domestic and foreign organisations to train and cover the ARN new challenges.

The training of personnel will be a permanent task for the next years.

4.5. SAFETY ASSESSMENT

ARN is presently continuing the safety assessments in significant issues, as it is common practice within the framework of supervision and safety review.

To this end, ARN has signed agreement with organisations as Purdue University to carry out independent accident assessment, based on development and application of a model of CNA II, using the programs Relap5 Mod3/Parcs (supported by USNRC).

Other assessment activities to cover for the next convention period are:

- *Integrity of the CNA I's RPV,*
- *Severe accident management program,*
- *PSA results and its applications,*
- *Periodical Safety Review applications for the License's renewals (including life extension).*

Many of these activities require a significant effort and need some years to be completed.

4.6. RISK-INFORMED APPROACH TO DECISION-MAKING IN THE REGULATORY BODY

Looking for a continuous improvement in its regulatory task, ARN is analyzing different tools in order to have a better use of its resources. In that order, in the framework of " Ibero-American Forum of Radiological and Nuclear Regulatory Agencies" Argentina become conscious that the Spain Regulatory body (CSN, Consejo de Seguridad Nuclear) has adopted the USNRC "Reactor Oversight Process" for the regulatory control of its NPPs.

To identified details about the system implementation, difficulties found and the effort needed, Argentina is going to participate in November 2007 in a specific meeting of the mentioned Forum, in order to analyze the possibility of implementing such system in ARN.

4.7. IMPROVEMENTS IN THE EMERGENCY PREPAREDNESS

An important modification of the Emergency Control Centre, located at ARN Headquarters, was started in order to improve the operative capacity, to be able to take advantage of the most modern technology and to upgrade the equipment.

In the near future it is foreseen to establish a point-to-point connection with the NPPs, to allow the on line transmission of information about the state of the plant and environmental measurements in emergency situations, and to permit videoconferencing in a more efficient way. Also, the update of the Geographical Information System and the capacity for transmitting information processed at the Emergency Control Centre through Internet with a secure connection is foreseen.

ANNEXES

ANNEX I

CONCLUSIONS ABOUT ARGENTINA DURING THE FIRST REVIEW MEETING ON THE CONVENTION ON NUCLEAR SAFETY

I.1. LEGISLATION AND REGULATORY FRAMEWORK

- Since 1994, Argentina has had an independent regulatory authority, in accordance with a national law. It is the single authority in charge of licensing and supervision of nuclear installations as well as personnel licensing. It is an independently financed from the national budget and regulatory fees.
- The regulatory system relies on performance-based regulation and is reliant on continuous interaction between regulator and licensee.

I.2. SAFETY OF NUCLEAR INSTALLATIONS

- A systematic operating experience feedback program and an ageing management program are in place.
- Accident management program is under development.
- Part of the major backfitting measures of Atucha I is completed, and the remaining backfitting measures are planned to conclude in 2001.
- The combined use of the PSA and deterministic approaches for regulatory decision making and to improve the operating conditions of the installations is considered a good practice.
- A continuous risk management program to improve safety using PSA, reassessment and evaluation of various options for improvement is in place.
- A periodic safety review is being performed every five years.
- Regulatory predictive performance indicators are used as a complementary preventive tool to detect early signs of deterioration.

Concern:

Although a large portion of the safety backfitting has been completed on CNA I, there still remain some important measures to be implemented.

Recommendation:

Argentina should expedite the backfitting program of CNA I in a timely manner.

I.3. SAFETY CULTURE/HUMAN FACTORS/ QUALITY ASSURANCE (MANAGEMENT OF SAFETY)

- A corporate policy and principles manual have been issued which refers to safety culture and the basis in which safety culture is cultivated.

I.4. RADIATION PROTECTION

- The legislative and regulatory framework in the area of radiation protection is in place.
- ICRP-60 recommendations for public and workers were implemented in 1995.

I.5. EMERGENCY PREPAREDNESS

- Emergency planning covering the on-site and off-site responses is in place, and periodic exercises are carried out on a regular basis.

Argentina provided and presented a very informative and comprehensive report and answered the questions in the same manner.

The participating Contracting Parties compliment the Argentine delegation for their excellent and informative presentation utilizing the latest visualization technology.

The participating Contracting Parties recognize Argentina's dedications to further improve the high level of safety of its nuclear installations and encourage a continuation of assessment and improvement of nuclear safety.

ANNEX II

CONCLUSIONS ABOUT ARGENTINA DURING THE SECOND REVIEW MEETING ON THE CONVENTION ON NUCLEAR SAFETY

II.1. INTRODUCTORY COMMENT

The presentation was well structured and addressed the improvements made since the 1st. Review Meeting.

Major Themes and Good Practices

There had been an aggressive backfitting program at Atucha I

- Requested by the regulator in 1998 with deadlines.
- Deadlines were not met and regulator refused to accept a revised schedule which had been proposed by the utility.
- The regulator took enforcement action in 2000.
- The plant was shutdown.
- Eventual restart was in 2001.
- All outstanding backfit tasks are to be completed during the 2002 outage.
- Since the original backfit program was established, additional tasks (reactor internals) have been identified and incorporated.

These demonstrate good regulatory practice in both requesting and enforcing safety requirements.

- There is a strong inspection program at Embalse to demonstrate that the plant will be capable of operating for its full design lifetime.
- A shutdown and low power PSA is being carried out at Embalse (completion 2003).
- Changes to the licensee organization which may impact safety must be submitted to the regulator before implementation.
- The regulator has established an oversight process for the systematic review of safety culture.
- Re-evaluation of siting (i.e. external hazards) has been carried out. Specifically: seismic events, tornadoes and flooding.
- There is a systematic OPEX program in place with a regulatory assessment of its effectiveness.
- De-regulation has had no observed impact on safety to ate.
- The regulator has full autonomy and adequate capability, but recently there has been a small reduction in staffing levels due to retirements.

II.2. POINTS FROM THE DISCUSSIONS

- A recent law requires that the regulator is responsible for decision-making and co-ordination of actions associated with emergency Planning and Preparedness. This is a new activity that is still being developed.
- The Members of Group V noted that the radiation dose levels during the necessary backfits at Atucha I were relatively high. The Members of Group V encourages the regulator to monitor these trends closely during the residual work.

- The regulator is developing Safety Performance Indicators. The Members of Group V recommended that thresholds be defined in these indicators which can be used as measures for acceptable safety performance and regulatory action.

II.3. POINTS FROM THE 1ST REVIEW MEETING NOT DISCUSSED ABOVE

The following information was provided during the presentations:

- A non-prescriptive strategy is preferred.
- There is currently no difficulty in providing financing for safety upgrades of the operating reactors.
- PSA is widely used; continuous review processes are in place as a substitute for PSR; safety reports are updated as part of this process.
- Emergency preparedness exercises are carried out once per year.

II.4. ITEMS TO BE INCLUDED IN THE NATIONAL REPORT FOR THE THIRD CNS REVIEW MEETING

The Members of Group V suggested that updates on the item given in D above should be included in the next report.

ANNEX III

CONCLUSIONS ABOUT ARGENTINA DURING THE THIRD REVIEW MEETING ON THE CONVENTION ON NUCLEAR SAFETY

III.1. HIGHLIGHTS

- ATUCHA 1: Implementation of an extensive backfiting programme prioritizing reactor internal issues, implementation of a second heat sink system, and analysis of RPV integrity.
- ATUCHA 2: completion of the plant construction licensed since 1981, may need additional requirements to incorporate latest improvements in nuclear field.
- Improvement in the areas of: safety culture, human factors, ALARA activities, Emergency Preparedness
- Following the decision to perform 10 – year periodic safety reviews the Regulator decided to grant operating licenses limited to 10 years
- Operating experience feedback:
  - Long list of actions taken based on national and international experiences
- Implementation of quality management system in the Regulatory Body
- Development of regulatory safety indicators
- More openness and transparency in Regulatory Body and Operator relationship.

III.2. 2nd REVIEW MEETING FOLLOW UP

Items emerged from the discussions:

- Emergency planning and preparedness activities
  - Rely on technical knowledge, strong legal framework, large dissemination of information, education and trainings.
- ALARA activities
  - Plant specific and headquarter (HQ) ALARA committees are established
  - Specific plant programme implemented in several areas led to significant doses reduction
- Regulatory safety indicators
  - 25 indicators defined using acceptable ranges rather than thresholds
  - Mainly based on operating experience
  - Used as additional regulatory tools

III.3. GOOD PRACTICES

- Strong programme on openness and transparency regarding community participation and general public information in the field of emergency preparedness and exercises on an annual basis required by law.
- Regulatory Body training programme based on profile of each post (please elaborate at the next CNS RM).

III.4. CHALLENGES

- Atucha 1: First stage of the sever accident management programme just finished.
  - Effectiveness of identified scenarios must be assessed to elaborate corresponding procedures and training programme.
- ATUCHA 2: Meeting current regulatory requirements for a 30 – year old design in compliance with international applicable standards to the extend practicable
- ATUCHA 2: Status of long time stored equipment: comprehensive programme envisaged to evaluate the situation, spare parts issue.
- Development of indirect safety culture indicators
- RB concerns regarding ATUCHA 1 RPV integrity before the end of life time
- RB human resources might be an issue by 2010 due to many retirements

III.5. PLANNED MEASURES TO IMPROVE SAFETY

- Continuation of the Severe Accident Management (SAM) programme implementation
  - Progress could be reported at he next CNS RM
- Plant specific seismic PSA programme in progress for Embalse plant (PSA was not considered in original design)
  - Progress could be reported at the next CNS RM
- Continuing implementation of quality management programme within the RB
- Application of PSA results at ATUCHA 1 and EMBALSE
- Regulatory audit scheduled in 2005 to improve operating experience feedback in ATUCHA 1
- Peer reviews planned in 2006 / 2007.

ANNEX IV

ANSWER TO QUESTIONS OR COMMENTS-

NATIONAL NUCLEAR SAFETY REPORT - 2004

N° 1
CNS-Ref.-Art.: 6
Page of Report: 16
Chapter of Nat. Report: 16.3

What Regulatory approach will be applied to a possible completion of Atucha II with regard to:
a) Monitoring of status of current construction and of components already on-site
b) Adaptation of older standards to current safety standards

Part a): The licensing process of NPPs considers 4 types of licenses: Construction, Commissioning, Operation and Decommissioning. The Atucha II NPP Construction License was granted by the Regulatory Body on July, 1981. At the beginning the construction activities were developed according the established schedule. However, since 1986 the construction was delayed and consequently the corresponding licensing process.

- Until now, the main regulatory activities have included:
- Preliminary Safety Analysis Report assessment.
- Preliminary Risk Analysis Report assessment.
- Responsible Organization, main suppliers, and main contractors audits.
- In-situ regulatory inspections during fabrication and installation of main components.
- Regulatory inspections to building site and components store.
- Mandatory documentation assessment.
- Brazilian/Argentinean regulatory bodies interaction to get experience from Angra II NPP licensing process, which had similar delay during the erection like the Atucha II NPP case.
- Safeguards design questionnaire report assessment.

It is important to mention that Atucha II has design safety features improved respect to Atucha I because it is similar to German 3rd and 4th generation plants and many aspects of Atucha I (unique German PHWR) operating experience were already considered in the Atucha II updated design such as:

- Significant operating events of Atucha I, in particular the most relevant event happened in 1988 “Fuel channel break”. Such event provoked reactor internal damages involving relevant corrective actions and design changes (see Annex 6, National Nuclear Safety Report 1998 “IAEA Safety Review Mission at Atucha I NPP”). This experience was considered to improve Atucha II original design.
- Reactor pressure vessel surveillance program.
- Reactor internal without “Stellite-6”.
- Emergency feedwater system (second heat sink for Atucha I) is already included in Atucha II original design.

However, there are additional features of Atucha II already considered coming from external experience mainly KWU reactors such as:

- TMI 2 designer implemented changes
- H2/D2 recombinators
- Improvements on computer operation support systems
- Updated safety analysis codes
- Environmental qualified of I&C, electrical and mechanical components and equipments

On the other hand, the Regulatory Body considers that it would be necessary to issue additional requirements due to last years improvements in the nuclear field associated with nuclear and radiological safety, safeguards and physical protection. The most relevant aspects to be considered are:

- Update of the Preliminary Safety Analysis Report in particular those issues regarding to plant risk assessment and compliance of the Standard AR 3.1.3 “ Radiological Criteria Related to Accidents in NPPs”.

- Update of external events analysis.
- Severe accidents assessment.
- Application of operating experience feedback from Atucha I.
- Detailed inspections on all components and equipments previously to their assembly.
- Update Safeguards documentation.
- Update Physical Protection Report.

There are some examples with delayed constructions of NPP around the world. In particular the Brazilian Angra II NPP, now in operation, can be mentioned due to its german technology like Atucha II, and delayed almost two decades according the original construction plan. The Argentinean Regulatory Body considered the Angra II licensing process to get the corresponding experience to be applied at Atucha II. There were no significant design changes in Angra II from the licensing process related to safety related components, equipments and systems. A detailed inspections on all components and equipments previously to their assembly was one of the main issues due to their long storage period exceeding the original schedule.

Part b): The main basis to grant the Atucha II Construction License were the considerations stated in the Preliminary Safety Analysis Report. Such report includes the verification of Standard AR 3.1.3 "Radiological Criteria Related to NPP Accidents". The range of accidents covered for this standard are not classified discriminating design basis and beyond design basis accidents but considering that any accident can potentially occur and they has two parameters associated, its occurrence probability and its consequence. The consequences are measured in terms of individual doses.

The Regulatory Body requires that the Safety Report must be updated according to the state of the art at the licensing time, if applicable. However, in all cases, based on AR 3.1.3 the overall plant risk must be acceptable. In this sense, such update must include: the plant safety characteristics, the operating experience feedback from Atucha I, the operating experience applicable from other plants and updating coming from regulatory requirements.

N° 2
CNS-Ref.-Art.: 6
Page of Report: 14
Chapter of Nat. Report: 6.1.5

According to the information given in the National Report, it appears that there exists no verified safety assessment of the RPV integrity of Atucha I. Please give more information on the justification for continued operation.

The RPV assessment carried out involves material analysis and testing for both irradiated and non-irradiated samples, samples lead factor, neutron spectra evaluations and pressurized thermal shock calculations (PTS).

Most of the assessments were performed contracting external services from KWU, FRAMATOME, National Atomic Energy Commission (CNEA) and international experts.

During 2002, the Responsible Organization organized a workshop among several technical support organizations such as Oak Ridge NL, GRS, VTT, FRAMATOME and CNEA to review the reactor pressure vessel status, in particular RPV material related issues. However, the Regulatory Body still has concerns about specific uncertainties resulting from the assessment . In this sense, it was required some additional efforts to guarantee the RPV integrity to the end of lifetime.

The main conclusion from the above mentioned activities was that the RPV integrity was assured far beyond the deadline to met the regulatory requirements and this was the basis for continued operation of Atucha I.

N° 3
CNS-Ref.-Art.: 6
Page of Report: 13
Chapter of Nat. Report: 6.1.5

Will there be any non-destructive testing of the critical areas of the RPV? If yes, How will this inspection be performed?

The RPV shell is a cylindrical assembly comprising two courses of rolled plate. The RPV shell material is low – alloy steel plate which is understood to be similar to ASME II SA508 Class II material. The plate is 220 mm thick except at the bottom end of the shell where it tapers to match the 120 mm thickness of the bottom dome.

During the next planned outage, March 2005, it is programmed to perform an automatic ultrasonic inspection of specific RPV welding. The following welds are included in the inspection's scope (see the attached figure):

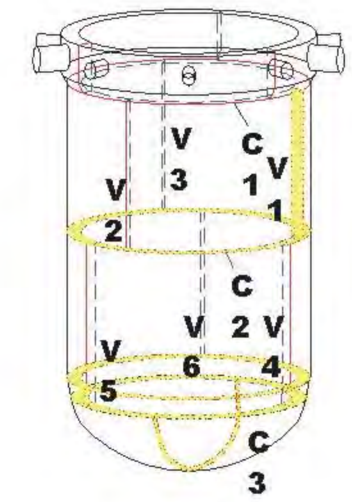
- C2, circumferential weld joining the two shell courses.
- V1, longitudinal weld in the upper shell course.
- V5 and V6, longitudinal weld in the lower shell course.

Also it will be performed the inspection of R3 and R4, Main Coolant System nozzle to shell welds.

Mitsui Babcock Energy Limited from Scotland will do the inspection, which is designed primarily to detect and size longitudinal and transverse defects of through wall orientation. The inspection will be to cover the examination volume defined by ASME CODE XI, IWB 2500-1 to the extent practicable.

Regarding outside inspections, a magnetic tractor capable of free movement over the vessel surface is launched onto the vessel surface. The tractor will carry an array of 3 UT probes with variable angle. Ultrasonic data will be acquired for each probe in the search array at a grid of points over the weld surface under control of the Nuclear Electric "MIPS" software.

Besides, NASA has already performed two additional inspections: during the planned outage of 1987, an inspection of longitudinal welding: V1, V5, V6 and circumferential welding: C1, C2 was done using a Rockwell system. The second one, was done in 1994 and the examinations were carried out by FORCE Institute at C2 and V2 welds.



N° 4
CNS-Ref.-Art.: 6
Page of Report: 13
Chapter of Nat. Report: 6.1.5

Which modifications were performed on the low-pressure injection system? And why were they performed?

The low pressure injection system design changes were carried out in the I&C logic to be compatible with second heat sink (emergency feedwater supply system) functions. The main change was related to the pressure set-point of small LOCA signal to avoid its actuation in case of specific secondary steam breaks and light water injection to the primary system which contains heavy water. Thermohydraulics analysis demonstrated that the new pressure set-point value must be 93 ate instead of 104 ate (original design value).

The following figure shows the primary pressure evolution corresponding to a 2A size steam break inside the containment, such pressure remains above 93 ate.

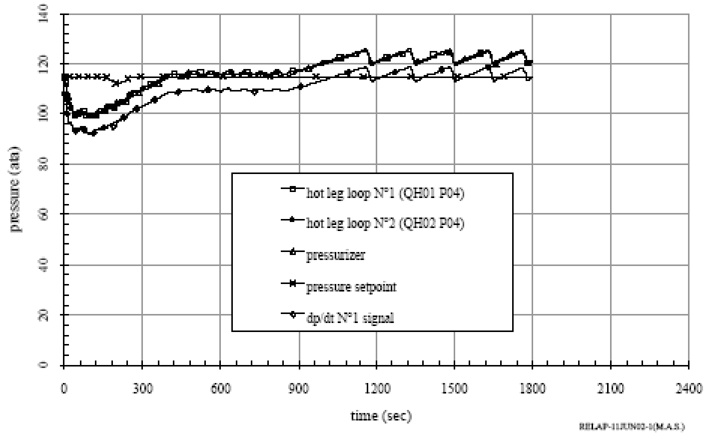


Figure N° 1: **Main steam line break (Normal Power Case). (2A) inside the containment. NZ52 by 90 ata. Primary pressure**

Another design change was concerning to the delay 15 minutes the sump flooding discharge tanks signal after LOCAs signals were triggered. It increases the water temperature by mixing the water coming from the primary break together with water discharged from the tanks to reduce pressurized thermal shock effect.

N° 5
CNS-Ref.-Art.: 6
Page of Report: 14
Chapter of Nat. Report: 6.1.5

The Section 6.1.5 under item b) states that "Additional measures were taken focused on reducing the impact of the thermal shock in particular reactor protection changes related to low pressure injection system". Provide some more information on measures, taken to reduce the impact of the thermal shock in particular reactor protection changes related to low pressure injection system.

The low pressure injection system design changes were carried out in the I&C logic to be compatible with second heat sink (emergency feedwater supply system) functions. The main change was related to the pressure set-point of small LOCA signal to avoid its actuation in case of specific secondary steam breaks and light water injection to the primary system which contains heavy water. Thermohydraulics analysis demonstrated that the new pressure set-point value must be 93 ate instead of 104 ate (original design value).

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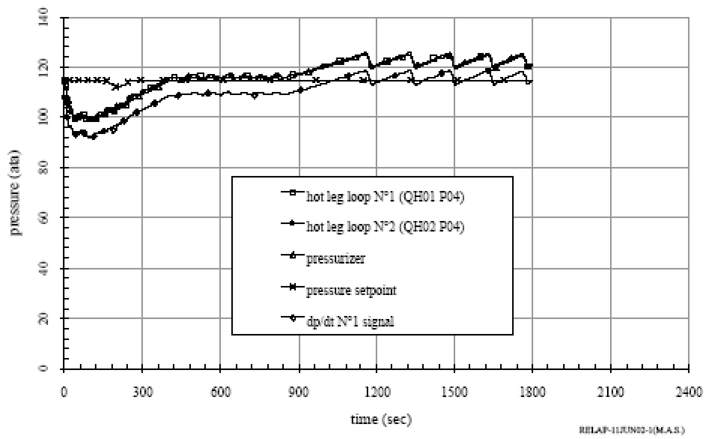

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Another design change was concerning to the delay 15 minutes the sump flooding discharge tanks signal after LOCAs signals were triggered. It increases the water temperature by mixing the water coming from the primary break together with water discharged from the tanks to reduce pressurized thermal shock effect.

N° 6
CNS-Ref.-Art.: 6
Page of Report: 13
Chapter of Nat. Report: 6.1.5

Current experience from LWRs states that heating up the emergency core cooling water (low- and high-pressure systems) is an effective measure to mitigate Pressurized Thermal Shocks (PTS) on the RPV wall.
Please explain why this is considered not to be effective for the high-pressure injection of Atucha I.

The High Pressure Injection System (HPIS) water heating up is generally an effective measure to face PTS effect. However, considering the specific Atucha I NPP response when the HPIS is actuated and the corresponding timing involved in this scenario, it was demonstrated that the PTS effect is not relevant. Therefore, it was not considered necessary to heat up the high pressure emergency core cooling water. The results of the Atucha I RPV - PTS studies, are summarized below:

- Temperatures profiles and stresses in RPV wall during cooling down process
During RPV cool down process the temperature gradient in the RPV wall reach a maximum. The Atucha I PTS analysis result indicates that maximum stresses (maximum K_I) are presents together with the maximum wall temperature gradient.

The fracto-mechanical limit condition (FLC), reached when the K_I and K_{IC} curves are in contact, is given after the maximum stresses are present and it depends on both stresses and temperatures values which surround the crack. In the tables I, II and III are showed the following parameters of interest corresponding to different size and locations of breaks (LOCAs).

Table I

Initiating Event Break size (cm2)	Maximum K_I occurrence time (sec.)	Time to reach $K_I = K_{IC}$ (surface) (sec.)	Temperature to reach $K_I = K_{IC}$ (surface) (^oC)	Time to reach $K_I = K_{IC}$ (crack) (sec.)	Temperature to reach $K_I = K_{IC}$ (crack) (^oC)
10 (HL)	2470.	-	-	3970.	72.
20 (HL)	1520.	2270.	65.	2270.	82.
50(HL)	1140.	1720.	65.	2470.	78.
100 (HL)	759.	1140.	72.	-	-
200 (HL)	759.	759.	75.	-	-
500 (HL)	503.	759.	73.	-	-
2A (HL)	555.	-	-	1250.	76.
500 (LS)	1140	-	-	2140.	132.
PV.	1720.	-	-	7720.	56.

HL: hot leg, LS: loop seal, PV: pressurizer valve

Table II

Initiating Event Break size (cm2)	Time when accumulators start injection (sec.)	Accumulators injection end (sec.)	Low pressure injection start (sec.)
10 (HL)	785.	2200.	2200.
20 (HL)	635.	1800.	1200.
50 (HL)	480.	1435.	880.
100 (HL)	286.	818.	482.
200 (HL)	153.3	425.	236.
500 (HL)	66.	205.5	97.
2A (HL)	21.8	100.	13./21.
500 (LS)	74.5	235.	130.
PV	1220.	4230.	4223.

HL: hot leg, LS: loop seal, PV: pressurizer valve

Table III

Initiating Event Break size (cm2)	RPV wall temperature when FLC is reached (°C)	RPV wall temperature when ACC empty (°C)	Sump water temperature when FLC is reached (°C)	Outlet moderator heat exchanger temperature when FLC is reached (°C)
10 (HL)	**72.**	**120.**	**64.6**	**35.**
20 (HL)	**65.**	**70.**	**64.**	**32.5**
50 (HL)	**65.**	**70.**	**71.**	**37.8**
100 (HL)	**72.**	**80.**	**69.5**	**37.3**
200 (HL)	**75.**	**100.**	**65.3**	**37.1**
500 (HL)	**73.**	**110.**	**70.**	**38.8**
2A (HL)	**76.**	**160.**	**84.**	**45.6**
500 (LS)	**132.**	**225.**	**74.5**	**42.**
PV	**56.**	**120.**	**64.3**	**24.4**

HL: hot leg, LS: loop seal, PV: pressurizer valve

According to the tables showed above it is concluded that the emergency core cooling system influence on the RPV wall cooling process depends on the break size.

- Emergency cooling systems behavior

Break size: Larger than 200 cm^2:

Table I and II shows that the accumulators actuation time is met before to reach K_I maximum, it means that the accumulator function has a secondary role considering the maximum RPV wall gradient and the time to reach FLC is even larger (Table I, contact K_I and K_{IC} curves columns 3 or 5). Additionally, the actuation interval of HPIS is very short as to develop a significant RPV wall gradient (Table III, column 2). The wall temperature reached when the accumulators are empty are higher than the temperature when FLC is reached due to the accumulators water is discharged before to reach FLC.

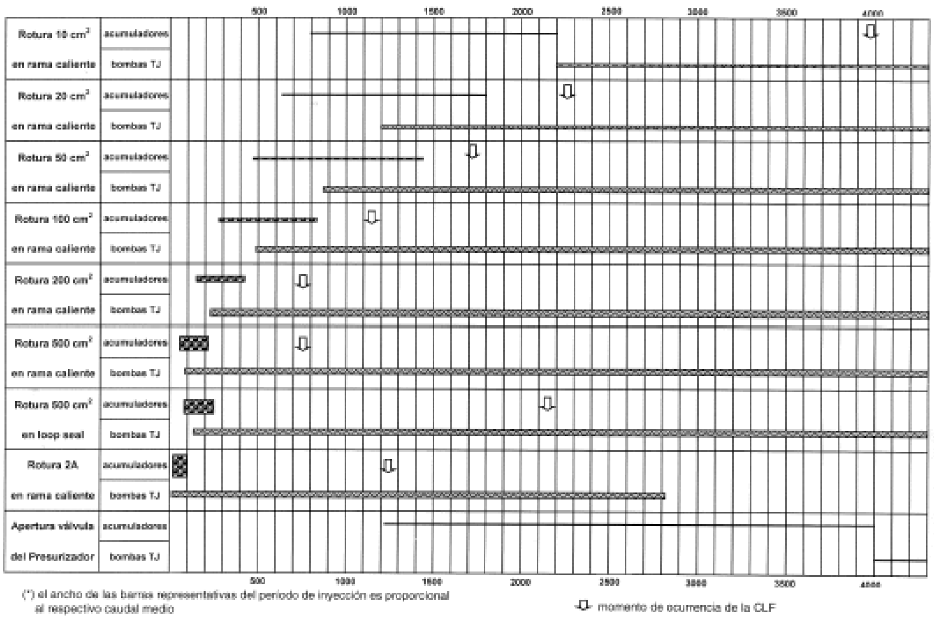
Break size: between 20 cm^2: and 200 cm^2:

The behavior corresponding to these break sizes is quite similar to those described before. However the time margins are more reduced to reach FLC. Additionally, the flow supplied by HPIS are smaller for this break sizes and this flow is mixed in the loop-seal together with the low pressure injection water. The heat-up of the HPIS water could be slightly more relevant in case of break sizes between 20 and 50 cm^2. However to face this situation, a design change was carried out to delay the sump flooding discharge tanks signal in 15 minutes after the LOCA signals were triggered. It increases the water temperature by mixing the water coming from the break with water discharged from the tanks to reduce the pressurized thermal shock effect.

Break size: lower than 20 cm^2:

The water injected from the HPIS is intermittent and it is stopped before to reach the maximum gradient and also far away to reach the FLC.

In conclusion, the benefits of heat-up the HPIS water are not relevant to face the PTS effects.



N° 7
CNS-Ref.-Art.: 6
Page of Report: 12
Chapter of Nat. Report: 6.1.2

It is mentioned that the licensing of such new safety system (second heat sink) involved an important effort to the Regulatory Body in terms of regulatory evaluations and inspections. In this regard, it may be elaborated as to what extent the overall plant safety has improved with the installation of this new system in term of core damage frequency? In addition, which safety standards were used for acceptance of this system by the regulatory body? How the power supply to this system is foreseen to be ensured in case of station blackout (loss of all off-site and on-site power except station batteries).

Part a): The safety impact considering the inclusion of the second heat sink as a special safety system contributes about approximately half order of magnitude in the overall core damage frequency.

Part b): The Second Heat Sink met the following standards:

1) Argentina Regulatory Standards:
   - AR 3.2.1 – General Safety Design Criteria for Nuclear Power Plants.
   - AR 3.2.3 – Fire Protection for Nuclear Power Plants.
   - AR 3.4.1 – I&C and Reactor Protection Systems for Nuclear Power Plants.
   - AR 3.5.1 – Emergency Power Supply for Nuclear Power Plants.
   - AR 3.6.1 – Quality System for Nuclear Power Plants.

Additionally, the German safety standard KTA-3503 "Type Testing of Electrical Modules for the Reactor Protection System" was applied to I&C digital technology such as the Teleperm XS System implemented. The qualification tests were performed by the Gessellschaft für Anlagen und Reaktor Sicherheit (GRS) together with the Technischer Überwachungs Verein (TÜV) from Germany. It also was qualified by NRC through tests performed by EPRI from United States. All mechanical components were designed and constructed according to KTA standards.

Part c)
One of the main functions of the Second Heat Sink is as Emergency Feedwater System that includes a separate, independent steam generator supply water feed system, feed from a dedicate water storage tank through two redundant lines, driven -electrical and mechanical- by exclusive and dedicated diesel engines. Such independent diesel engines allow to ensure the power supply in case of station blackout occurrence.

N° 8
CNS-Ref.-Art.: 6
Page of Report: 13
Chapter of Nat. Report: 6.1.4

Section 6.1.4 of the Report mentions the so far achieved total core damage frequency (CDF) for Atucha I which is below 1x10-4/year, and in Section 6.2.3 it is stated that core damage frequency for Embalse NPP is 2.67×10^{-5}/year.
What is the main cause of the difference in CDF values for Atucha I and Embalse plants?.

The PSAs for both Atucha I and Embalse NPPs performed to quantify the plant safety in terms of core damage frequency are "plant-specific" PSAs. It means that any design difference with another plant can be seen in the modelling and quantification results. Additionally, Atucha I and Embalse NPPs belong to different design generations, so the design safety criteria is different among them.

On the other hand, plant damage states for Atucha I NPP were considered in a more conservative approach than in Embalse because the concept "core damage" used in the Atucha I PSA includes all the non-controlled reactor states, without making any difference between different final states of core damage, or the time since the beginning of the non controlled state up to the beginning of core damage.

However, a more detailed modelling of sequences using both better codes and an improved nodalization recently performed shown that some plant damage states initially considered in the Atucha I PSA were very conservative, in particular those related to the operator time-windows to take corrective actions. This new development will allow to update the Atucha I PSA and obtain new results that will impact in the core damage frequency.

N° 9
CNS-Ref.-Art.: 6
Page of Report: 15
Chapter of Nat. Report: 6.2.2

With reference to section 6.2.2, it is concluded that "the results in the evolution of the inspection that took place allow a life expectancy higher than 25 years for all sensitive areas". What are the bases for this conclusion regarding life expectancy of Embalse Feeders?

The conclusion stated in the paragraph 6.2.2 is based on the inspection results. In the case of flow accelerated corrosion (erosion-corrosion) damage, the ultrasonic inspection results showed that the feeders average thickness is similar to the obtained in the previous inspections. So, the thickness decreasing rate is practically constant showing that there is no damage mechanism acceleration.

Similar conclusions were stated in the case of stress corrosion cracking. In spite of a more reliable (regarding previous methods) crack detecting and sizing methodology was used (according to COG procedure), it could be concluded that there are not relevant indications in whole feeders.

N° 10
CNS-Ref.-Art.: 6
Page of Report: 14
Chapter of Nat. Report: 6.1.6

It has been mentioned that after the increased drop time and sticking of Shutoff Control Rods in their guide bushing, whether plant was allowed to operate with this problem? Was back-fitting to replace all of them required by the regulatory body before proceeding to plant power operation or power operation allowed conditionally under an exemption?.

During a shutoff control rod drop test an increase in the drop time of some shutoff control rods was verified. Therefore, the Regulatory Body required to the Responsible Organization to evaluate the actual control rods performance, carrying out several additional tests, inspections and analysis (see National Nuclear Safety Report, 2001, pages 16-17). As conclusion the shutdown system met its function within the technical specifications. On the other hand, its design is very conservative in terms of the reactivity shutdown margin, as well as the number of necessary rods for the reactor safe shutdown considering all potential accident scenarios. The above mentioned were the regulatory basis to allow to operate the plant but an upgrading program to replace all control rod guide tubes (as a part of the backfitting program) were required as a preventive measure.

This program was implemented in three planned outages by imposing a set of conditions corresponding to each plant start-up. In particular, specific drop tests and an inspection program focused on detecting any eventual bending and interactions between control rod guide tubes and coolant channels were carried out.

N° 11
CNS-Ref.-Art.: 6
Page of Report: 15
Chapter of Nat. Report: 6.2

For the Embalse NPP, Argentina implemented a new procedure to monitor deuterium uptake in pressure tubes by scraping the pressure tubes during their outage in May 2004. The report indicates that the results were not available at the time of completing the report.
Please provide details of the results of scraping the pressure tubes. Please indicate what changes will be made (if any) consequent to these results.

In the last planned shutdown a pressure tube campaign was conducted and ten tubes were selected for scraping, three of them were scraped also in 1998.

The samples have been analyzed at AECL laboratories in Canada.

Deuterium pickup was determined for five positions in each of the tubes considered and raw data is available. However, final conclusions on the pickup rate depend on the assessment of these results together with data from other CANDU plants, which provide sufficient statistics. The report will be available in April 2005.

N° 12
CNS-Ref.-Art.: 6
Page of Report: 15
Chapter of Nat. Report: 6.2.3

The report says that the CDF for Embalse includes containment bypass sequences. What is the contribution of core damage with containment bypass to the overall CDF?.
What would be the source term in this case?. Which recovery actions have been considered and implemented for this case?.

Containment by-pass sequences were considered in the Level I PSA for Embalse NPP. The contribution of this sequences to the core damage frequency is 8.6% without recovery actions.

Source term was not calculated for this sequence because is out of the scope of Level I PSA.

N° 13
CNS-Ref.-Art.: 6
Page of Report: 16
Chapter of Nat. Report: 6.3

For Atucha II NPP, the report indicates that "the plant construction activities are still almost discontinued".
Please explain what is meant by "still almost discontinued".

The wording of "still almost discontinued" means that certain activities were ongoing but in a very slow manner. During the considered period the activities carried out were:

- Electromechanical components preservation.
- Maintenance of some systems such as Emergency Power Supply system including the diesels generators.
- Technical support of certain remaining activities.
- Some additional technical specifications to be used by future service contracts
- Updated analysis including technical, economical and financial aspects for the Government decision making process to finalize the plant construction

The Atucha II NPP Construction License was granted by the Regulatory Body on July,1981. At the beginning the construction activities were developed according the established schedule. However, since 1986 the construction was delayed and consequently the corresponding licensing process. The construction activities have never been discontinued, maintaining different construction intensity.

N° 14
CNS-Ref.-Art.: 6
Page of Report: 11
Chapter of Nat. Report: 6

Which are the existing nuclear installations under the Convention in Argentina? Please provide for an exhaustive list.

As was stated in the National Nuclear Safety Reports, the country has two nuclear power plants in operation, Atucha I and Embalse, which initiated their commercial operation in 1974 and 1984 respectively. Their corresponding net electric power are 335 MW and 600 MW, which represent about 7.3% of the installed electric power. Both nuclear power plants supply about 12% of the total electric power generated in the country. A third nuclear power plant, Atucha II, is in an advanced state of construction.

Atucha I NPP is located about 100 km to the Northwest of Buenos Aires City. The reactor is of the PHWR type with a pressure vessel. According to the original design Atucha I is fuelled with natural uranium, but fuel elements of new design were incorporated with slightly enriched uranium (0.85%). The reactor is moderated and cooled with heavy water (see Annex 8 NNSR 1998).

Embalse NPP is located at the Province of Córdoba, approximately 110 km to the south of Cordoba City (state capital) and 5 km to the Southwest of Embalse town. It is a CANDU type reactor, of the pressure tube type, loaded with natural uranium fuel and moderated and cooled with heavy water (see Annex 8 NNSR 1998).

Atucha II NPP is under construction, next to Atucha I. Its reactor will also be of the PHWR type with a pressure vessel, loaded with natural uranium fuel, moderated and cooled with heavy water, with a net electric power of 693 MW.

N° 15
CNS-Ref.-Art.: 7
Page of Report: 19
Chapter of Nat. Report: 7.3

Section 7.3 of the Report says that the nuclear installations' Safety Reports have to be updated every 5 years. However, in 2003 the Regulatory Body of Argentina took a decision on performing the regular Safety Review for each NPP when issuing an operation license.

What is the adopted frequency of performing NPP Safety Reviews if the term of operation license is limited by 10 years?

Since the beginning of the Argentinean NPPs operation in 1974, it was established that the Operating Licenses were not limited in time but, its validity were based on the fulfilment of the licensing conditions, regulatory standards and specific regulatory requirements. The Regulatory Body applies a routine program to verify the nuclear and radiological safety conditions through inspections, audits and safety assessments.

However, in 2003 the Regulatory Body decided to initiate a systematic safety reassessment (periodic safety review – PSR-) every 10 years, based on the IAEA Safety Standard NS-G-2.10 PSR of NPPs, to assess the cumulative effects of plant ageing and plant modifications, operating experience, technical developments and siting aspects. The reviews include an assessment of plant design and operation against current safety standards and practices, and they have the objective of ensuring a high level of safety throughout the plant's operating lifetime. They are complementary to the routine and special safety reviews and do not replace them.

Based on PSR, the Regulatory Body decided to limit the validation time of the Operating Licenses to 10 years and such PSR results will be used to determine the Operating License renewal.

On the other hand, the update period of the Safety Reports remain unchanged as stated in the regulatory standards that establish to update such reports at least every 5 years or in case of significant safety plant modifications.

N° 16
CNS-Ref.-Art.: 7
Page of Report: 19
Chapter of Nat. Report: 7.3

It is understood that formal periodic safety reviews are required by new operating licenses. What was the basis for granting the recent ten-year operating license? Has the safety analysis report been updated?

Since the beginning of the Argentinean NPPs operation in 1974, it was established that the Operating Licenses were not limited in time but, its validity were based on the fulfilment of the licensing conditions, regulatory standards and specific regulatory requirements. The Regulatory Body applies a routine program to verify the nuclear and radiological safety conditions through inspections, audits and safety assessments.

However, in 2003 the Regulatory Body decided to initiate a systematic safety reassessment (periodic safety review – PSR-) every 10 years, based on the IAEA Safety Standard NS-G-2.10 PSR of NPPs, to assess the cumulative effects of plant ageing and plant modifications, operating experience, technical developments and siting aspects. The reviews include an assessment of plant design and operation against current safety standards and practices, and they have the objective of ensuring a high level of safety throughout the plant's operating lifetime. They are complementary to the routine and special safety reviews and do not replace them.

Based on PSR, the Regulatory Body decided to limit the validation time of the Operating Licenses to 10 years and such PSR results will be used to determine the Operating License renewal.

On the other hand, the update period of the Safety Reports remain unchanged as stated in the regulatory standards that establish to update such reports at least every 5 years or in case of significant safety plant modifications.

N° 17
CNS-Ref.-Art.: 7
Page of Report: 19, 103
Chapter of Nat. Report: 7.3

What are the results for low-power and shutdown states in the PSA and which design changes have been required and implemented in the context of granting the recent operating licenses?

Atucha I, main conclusion of shutdown PSA was that shutdown risk is in the order of the risk corresponding to normal operation. The outage time for maintenance and plant configuration control resulted critical for specific plant components such as:

- Main pumps of residual heat removal system and related operator actions.
- Moderator pumps combined with the failure of residual heat removal system.
- Residual heat removal valves failures.

The main plant changes were related to the configuration control about the above mentioned components. It was not necessary to issue regulatory requirements about this subject.

Embalse shutdown and power PSA is in progress (deadline 2005).

N° 18
CNS-Ref.-Art.: 8
Page of Report: 23
Chapter of Nat. Report: 8.3

It should be stated clearly whether the Regulatory Body is already certified for ISO 9000:2001 or if another Quality Management System will be implemented in 2005.

The Regulatory Body is not certified for ISO 9000:2000. The Regulatory Body structured its quality management system based on ISO 9000:2000 and IAEA PDRP-6. However, it was decided to certify the following subsystems under ISO Standards:

- Regulatory Body Laboratories.
- Post-Graduate Nuclear and Radiological Course managed by Regulatory Body.
- Transport of radioactive materials system.

N° 19
CNS-Ref.-Art.: 8
Page of Report: 23
Chapter of Nat. Report: 8.3

It is appreciated that considerable detail has been provided regarding the quality management system within the regulatory body. Could Argentina elaborate, whether, the system is subjected to any external/international audit?

The Quality Management System implementation is in progress and it will be subjected to conduct internal and external audits. International audits will also be considered.

N° 20
CNS-Ref.-Art.: 8
Page of Report: 23
Chapter of Nat. Report: 8.3

Australia notes the activities being undertaken to strengthen Quality Management within the Regulatory Body based on ISO 9000:2000 and IAEA PDRP-6 'Quality Management in Regulatory

Bodies'. Will the new Quality Management arrangements be certified and its effectiveness monitored by an external certification organization? If so, will this surveillance address both ISO 9000 based requirements and those of IAEA PDRP-6?

The Regulatory Body is not certified for ISO 9000:2000. The Regulatory Body structured its quality management system based on ISO 9000: 2000 and IAEA PDRP-6. However, it was decided to certify the following subsystems under ISO Standards:

- Regulatory Body Laboratories.
- Post-Graduate Nuclear and Radiological Course managed by Regulatory Body.
- Transport of radioactive materials system.

The Quality Management System implementation is in progress and it will be subjected to conduct internal and external audits. International audits will also be considered.

N° 21
CNS-Ref.-Art.: 8
Page of Report: 24
Chapter of Nat. Report: 8.3.1

Clarification is sought that the total of 120 hours training is provided to each staff member in the positions identified in Section 8.3.1 of the report?

The Regulatory Body carried out different training courses for a total amount of 120 training hours. 87% of the Regulatory Body personnel received an average of 20 training-hours.

N° 22
CNS-Ref.-Art.: 8
Page of Report: 25
Chapter of Nat. Report: 8.4

In the section on the maintenance of the level of competence of the Regulatory Authority, the Report states as follows: "To maintain and improve the regulatory quality and efficiency; in particular to address the challenge of the gradual loss of specialized human resources with the scientific and technical knowledge required to guarantee the quality of the regulatory decisions and of the control activities". How will this strategic goal be addressed through specific actions? Could some information on this issue be provided?

It is important to mention that due to the socioeconomic situation in Argentina, the lost of specialized human resources has been very low since the last ten years. As a consequence, it is estimated that the situation would be critical after the year 2010 due to amount of retirements.

However, the Regulatory Body has initiated specific actions considering this issue since 3 years ago. In this sense, there was an agreement with the National Executive Power that let specific actions resulting in an increase of financial and human resources to compensate the gradual loss of specialized human resources with the scientific and technical knowledge. Therefore, new human resources have been incorporated to the Regulatory Body and they are in a specific training program to foresee gradually the above mentioned issue consequences.

Examples of specific actions in progress:

- Additional resources to increase the annual budget to provide human resources.
- Quality management system implementation focused on the nuclear stakeholders and the continuous improvement concept.
- Professional career improvements.
- Post profile definition.
- Improved annual training program.
- Strategic plan of internal and external communication.

These actions in progress are in different implementation stages and it will be carried out approximately within the next five years.

N° 23
CNS-Ref.-Art.: 8
Page of Report: 25
Chapter of Nat. Report: 8.4

Please provide more details on the approach to maintain the competence of the regulatory body under the National Plan for the Modernization of the Public Sector.

It is important to mention that due to the socioeconomic situation in Argentine, the lost of specialized human resources has been very low since the last ten years. As a consequence, it is estimated that the situation would be critical after the year 2010 due to amount of retirements and organization ageing.

However, the Regulatory body has initiated specific actions considering this issue since 3 years ago. In this sense, there was an agreement with the National Executive Power that let specific actions resulting in an increase of financial and human resources to compensate the gradual loss of specialized human resources with the scientific and technical knowledge. Therefore, new human resources have been incorporated to the Regulatory Body and they are in a specific training program to foresee gradually the above mentioned issue consequences.

Examples of actions in progress:

- Additional resources to increase the annual budget to provide human resources
- Quality management system implementation focused on the nuclear stakeholders and the continuous improvement concept.
- Professional career improvements
- Post profile definition
- Improved annual training program
- Strategic plan of internal and external communication

Those actions in progress are in different implementation stages and it will be carried out approximately within the next five years.

N° 24
CNS-Ref.-Art.: 8
Page of Report: 25
Chapter of Nat. Report: 8.4

Addressing the “Program Agreement” between the ARN and the Ministers’ Chief Cabinet, the report states that “under this agreement, the ARN undertakes to improve its work”.
Please provide examples as to how the ARN has used this “Program Agreement” to maintain and enhance the competence of its staff.

It is important to mention that due to the socioeconomic situation in Argentina, the lost of specialized human resources has been very low since the last ten years. As a consequence, it is estimated that the situation would be critical after the year 2010 due to amount of retirements.

However, the Regulatory Body has initiated specific actions considering this issue since 3 years ago. In this sense, there was an agreement with the National Executive Power that let specific actions resulting in an increase of financial and human resources to compensate the gradual loss of specialized human resources with the scientific and technical knowledge. Therefore, new human resources have been incorporated to the Regulatory Body and they are in a specific training program to foresee gradually the above mentioned issue consequences.

Examples of actions in progress:

- Additional resources to increase the annual budget to provide human resources.
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- Professional career improvements.
- Post profile definition.
- Improved annual training program.
- Strategic plan of internal and external communication.

Those actions in progress are in different implementation stages and it will be carried out approximately within the next five years.

N° 25
CNS-Ref.-Art.: 8
Page of Report: 21, 25
Chapter of Nat. Report: 8.2, 8.4

Please update us on the issue of human resources. This was identified at the last CNS meeting as an issue worldwide. Your report states in Section 8.2 you have been hiring professionals, and Section 8.4, page 25, states there is a challenge regarding the gradual loss of staff with the technical knowledge required to guarantee the quality of the regulatory decisions and control activities. Is staffing an issue, and if so, how are you resolving it?

It is important to mention that due to the socioeconomic situation in Argentine, the lost of specialized human resources has been very low since the last ten years. As consequence, it is estimated that the situation would be critical after the year 2010 due to amount of retirements.

However, the Regulatory Body has initiated specific actions considering this issue since 3 years ago. In this sense, there was an agreement with the National Executive Power that let specific actions resulting in an increase of financial and human resources to compensate the gradual loss of specialized human resources with the scientific and technical knowledge. Therefore, new human resources have been incorporated to the Regulatory Body and they are in a specific training program to foresee gradually the above mentioned issue consequences.

Examples of specific actions in progress:

- Additional resources to increase the annual budget to provide human resources.
- Quality management system implementation focused on the nuclear stakeholders and the continuous improvement concept.
- Professional career improvements.
- Post profile definition.
- Improved annual training program.
- Strategic plan of internal and external communication.

Those actions in progress are in different implementation stages and it will be carried out approximately within the next five years

N° 26
CNS-Ref.-Art.: 8
Page of Report: 21
Chapter of Nat. Report: 8.2

On 2002, the Board of Directors of Nuclear Regulatory Authority has been renewed. It is not very clear if ARN became independent from politic regime and if ARN uses external TSO to complete its level of expertise in the nuclear field. Could ARN provide additional information in that sense?

Part a): Act No 24804, "National Law of Nuclear Activity" (1997), sets that the Regulatory Body is in charge of the regulation and surveillance of nuclear activity concerning radiological and nuclear safety, physical protection and safeguards. It also establishes that the Regulatory Body has autarchy and complete legal capability to act in the field of private and public rights, and that its resources are basically integrated with regulatory fees and with State support. According to the provisions in Articles 17 and 18 of Act No 24804, the Regulatory Body is managed and administrated by a Board of Directors constituted by six members, one of them being the President. All members of the Board are appointed by the Executive Power. Additionally, the National Law of Nuclear Activity establishes that all members of the Board of Directors must demonstrate a proven knowledge and background in the nuclear field.

As a consequence of the economical and political crisis occurred in 2001, the National Government decided to establish some changes to re-structure the State Organization and it was decided to reduce the number of directors of the main State Departments. Regarding the Regulatory Body, the Board of Directors was reduced to three members instead of six. To carry out such changes it was initiated a new selection process of the three Board of Directors members that concluded in 2002 when the new national authorities had been elected. It is important to point out that one Director remain in the same position from the last six years (since 1999) and all of them are recognized senior experts in the nuclear field.

Part b): To perform its regulatory functions, the Regulatory Body 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. The relationship between the Regulatory Body and a wide variety of national and international TSOs is carried out through agreements which rule the cooperation provided by such institutions of acknowledged technical-scientific level and independent criteria (a detailed list about the relationship with other organizations can be observed in all National Nuclear Safety Reports, Article 8).

National Nuclear Safety Report 1998: Tables 8.4 and 8.5
National Nuclear Safety Report 2001: Tables 8.6.1; 2; 3 and 4
National Nuclear Safety Report 2004: Tables 8.5.1 and 8.5.2

N° 27
CNS-Ref.-Art.: 8
Page of Report: 23
Chapter of Nat. Report: 8.3

Please provide a list of examples of the mentioned 27 regulatory procedures and instructions of the QA documentation.

BOARD OF DIRECTORS		
PG-DIR-002	Rev. 00	DEVELOPMENT AND REVIEW OF STANDARDS AND REGULATORY GUIDELINES
BOARD OF DIRECTORS TECHNICAL SECRETARÍAT		
PP-SGPF-001	Rev. 00	PHYSICAL PROTECTION INSPECTIONS
PP-SGS-001	Rev. 00	NATIONAL SAFEGUARD INSPECTIONS
SCIENTIFIC SUPPORT DEPARTMENT		
PP-SGEA-001	Rev. 00	SAMPLES MANAGEMENT
PP-SGEFR-001	Rev 00	SAMPLES PROCESSING INFORMATION MANAGEMENT
NUCLEAR AND RADIOLOGICAL SAFETY DEPARTMENT		
PP-GSRN-001	Rev. 00	INES: RECORD, ASSESSMENT AND COMUNICATION
PP-SGRN-001	Rev. 00	NUCLEAR POWER REACTOR AUDITS
PP-SGRN-002	Rev. 00	REGULATORY INSPECTIONS AT NUCLEAR POWER PLANT DURING OPERATION
PP-SGRN-003	Rev. 00	REGULATORY INSPECTIONS DURING PLANNED OUTAGES AT NUCLEAR POWER PLANTS
PP-SGRN-004	Rev. 00	MANAGEMENT OF RELEVANT EVENTS AT NUCLEAR REACTORS
PP-SGRN-005	Rev. 00	MANAGEMENT OF INTERNAL EVENTS AND OPERATING EXPERIENCE IN NUCLEAR POWER PLANTS
PP-SGRN-006	Rev. 00	INSPECTIONS AND TECHNICAL REVIEW FOR RESEARCH REACTOR IN OPERATION
PP-SGRN-007	Rev. 00	REGULATORY ACTIVITIES RELATED TO RESEARCH REACTORS AND CRITICAL ASSEMBLIES EMERGENCY PLANS
PP-SGRN-008	Rev. 00	REGULATORY ACTIVITIES RELATED TO NUCLEAR POWER PLANTS INTERNAL EMERGENCY PLANS
PP-SGRN-009	Rev. 00	RESEARCH REACTOR AND CRITICAL ASSEMBLIES ASSESSMENTS
PP-SGRN-010	Rev. 00	NUCLEAR POWER PLANTS ASSESSMENTS
PP-SGRN-011	Rev. 00	NON ROUTINARY NUCLEAR REACTORS ASSESSMENTS
PP-SGDR-001	Rev. 00	IRRADIATION PLANTS INSPECTIONS
PP-TMR-001	Rev. 00	ASSESSMENT AND APPLICATION FORM FOR RADIOACTIVE MATERIALS TRANSPORT
PG-GSRN-002	Rev. 00	REGULATORY REQUERIMENTS CONTROL AT CLASS I FACILITIES
PG-GSRN-003	Rev. 01	OPERATING LICENCE MANAGEMENT FOR CLASS II FACILITIES
PG-GSRN-004	Rev. 00	PERSONNEL LICENSING OF CLASS I FACILITIES
PG-GSRN-005	Rev. 01	REQUEST FOR CLASS III FACILITIES REGISTER
PG-GSRN-006	Rev. 03	INDIVIDUAL AUTHORIZATION MANAGEMENT FOR CLASS II FACILITIES PERSONNEL
PG-GSRN-007	Rev. 00	CERTIFICATE OF APPROVALISSUE FOR RADIOACTIVE MATERIALS TRANSPORT

PG-GSRN-008	Rev. 01	AUTHORIZATION TO IMPORT AND EXPORT RADIOACTIVE MATERIALS
PG-GSRN-009	Rev. 01	AUTHORIZATIONS TO REMOVE / REINSTALL RADIACTIVE SOURCES
PG-GSRN-010	Rev. 01	ISSUING NOTIFICATIONS AND GENERIC RECOMENDATIONS TO ORGANIZATIONS THAT OPERATE CLASS II & III FACILITIES
PG-GSRN-011	Rev. 01	CLASS II & III FACILITIES INSPECTIONS
PG-GSRN-012	Rev. 01	ISSUING REQUERIMENTS FOR CLASS II & III FACILITIES
PG-GSRN-013	Rev. 01	INSPECTION DURING TRANSFER OF RADIACTIVE SOURCES
PG-GSRN-014	Rev. 02	CLASS II & III FACILITIES CLOSING AND OPERATING LICENCES SUSPENSION
PG-GSRN-015	Rev. 00	SHIPMENT AND CONSIGNMENT INSPECTION OF RADIOACTIVE MATERIALS
PG-GSRN-016	Rev. 00	ISSUING OF AUTHORIZATIONS FOR NON ROUTINARY PRACTICES
PG-GSRN-017	Rev. 01	MANAGEMENT OF INDIVIDUAL LICENCES TO WORK AT REGISTERED FACILITIES
IT-GSRN-001	Rev. 00	DOSE EVALUATION
IT-SGRN-001	Rev. 00	COMMUNICATION OF RELEVANT EVENTS IN DOMESTIC NUCLEAR REACTORS
IT-SGIRFR-001	Rev. 01	DOCUMENTS MANAGEMENT OF SGIRFR
IT-TMR-001	Rev. 00	REQUERIMENTS VERIFICATION BEFORE THE FIRST SHIPMENT
IT-TMR-002	Rev. 00	INSPECTIONS OF PACKAGE FUNCTIONAL TESTS
IT-TMR-003	Rev. 00	INSPECTIONS OF SPECIAL FORM RADIACTIVE MATERIALS TESTS
DEPARTMENT OF ADMINISTRATION		
PP-SGGEF-003	Rev. 00	BUDGET DISTRIBUTION AND CHANGES
PP-SGGEF-004	Rev. 00	PROGRAMMING, DISTRIBUTION AND MODIFICATION OF THE INCOMING BUDGET
UNIT OF PLANNING AND PROSPECTIVE		
PP-UPP-001	Rev. 00	WORK PLAN AND BUDGET DEVELOPMENT
PP-UPP-002	Rev. 00	WORK-PLAN AND BUDGET CONTROL
IT-UPP-001	Rev. 00	FELLOWSHIPS PROGRAMMING

N° 28
CNS-Ref.-Art.: 8
Page of Report: 21-22
Chapter of Nat. Report: 8.2

What is the current number of staff for the different basic functions of the regulatory body?

Currently, the Regulatory Body has 191 persons as permanent staff, 24 as contracted professionals and 28 as trainees. Total 243 persons.

- 2% support personnel
- 10% administration
- 20% technician
- 68% Specialized professional

Regarding the Safety of NPPs and Research Reactors:

- Resident inspectors: 4
- Professionals full time: 22
- Professionals part time (technical support): 38

N° 29
CNS-Ref.-Art.: 9
Page of Report: 29
Chapter of Nat. Report: 9

Are organizational and management aspects of the NPP licensees subject to review and acceptance by the regulatory body? If so, to which extent? Are the organizational structure and/or the licensee's quality management included in the periodic safety reviews?

The NPP licensee organization must be approved by the Regulatory Body. Additionally, any NPP organizational changes that could affect the licensee's capabilities to met its obligations and responsibilities must also be approved by the Regulatory Body before its implementation.

On the other hand, as a part of the regulatory personnel licensing system, the management positions of NPP´s organization that are involved with safety related decisions making are licensed by the Regulatory Body.

The organizational aspects are included in the Periodic Safety Review required to the plants as it is stated in the IAEA Safety Guide NS-G-2.10 "Periodic Safety Review of NPPs". Such guide was adopted by the Regulatory Body as a reference document to perform the PSR. In particular, the subject review of organization and administration included in the PSR is carried out to determine whether the organization and administration are adequate for the safe operation of the nuclear power plant. Additionally, quality assurance is not considered to be a separate safety factor because it should be an integral part of every activity affecting safety.

N° 30
CNS-Ref.-Art.: 10
Page of Report: 31
Chapter of Nat. Report: 10

What indirect safety culture indicators will be adopted for the assessment of safety culture status? Do you make comparisons of safety culture status at different plants/units and do you make an assessment of safety culture status?

As was stated in the National Report, article 10, the development of a set of indirect safety culture indicators are in progress. The first approach to this subject was to select from the current safety indicators list those indicators that indirectly are in connection with safety culture attitudes. In this selection are included Indicators related with:

Training:

- Number of hours devoted to training on safety-related issues.

Feedback from Operational Experience:

- Number of documented event analyses, findings or design modifications in similar power plants.

Internal Control:

- Number of internal technical audits.

Compliance with Regulatory Authority standards

- Number of pending Regulatory Requirements.
- Number of violations to the Mandatory Documents.

Abnormal Operation:

- Number of relevant events. When direct or root causes are associated with human deficiencies
- Safety Systems actuation's. Considering here only the ones related with human failures

With the advance of the project possibly more indirect indicators would be included, such as indirect indicator related with human performance indicators, regulatory audits findings and resident inspectors findings.

In addition, recognizing the complex subject, Argentina has a close follow up on the international development in this area.

The project is in progress, therefore, at the moment did not include comparisons among indirect safety culture indicators.

N° 31
CNS-Ref.-Art.: 10
Page of Report: 31
Chapter of Nat. Report: 10

Please provide more information on the Indirect Safety Culture Indicators and on the regulators training to deal with organisational factors? How are these activities integrated into the regulatory oversight process?

1) As was stated in the National Report, article 10, the development of a set of indirect safety culture indicators are in progress. The first approach to this subject was to select from the current safety indicators list those indicators that indirectly are in connection with safety culture attitudes. In this selection are included Indicators related with:

Training:

- Number of hours devoted to training on safety-related issues.

Feedback from Operational Experience:

- Number of documented event analyses, findings or design modifications in similar power plants.

Internal Control:

- Number of internal technical audits.

Compliance with Regulatory Authority standards

- Number of pending Regulatory Requirements.
- Number of violations to the Mandatory Documents.

Abnormal Operation

- Number of relevant events. When direct or root causes are associated with human deficiencies
- Safety Systems actuation's. Considering here only the ones related with human failures

With the advance of the project possibly more indirect indicators would be included, such as indirect indicator related with human performance indicators, regulatory audits findings and resident inspectors findings.

In addition, recognizing the complex subject, Argentina has a close follow up on the international development in this area.

1) The international information in this field is analyzed by a senior regulatory officer that interact with another specialists worldwide. Additionally, this specialist participated in several meetings related to this subject.
2) The set of safety performance indicators is used as a regulatory tool to provide an additional view of the nuclear power plants performance allowing to improve the ability to detect any eventual degradation on safety related areas. Once the indirect safety culture indicators program was completed these indicators will be added to the safety performance indicators in use.

N° 32
CNS-REF.-ART.: 10
PAGE OF REPORT: 31
CHAPTER OF NAT. REPORT: 10

The report mentions progress "such as development of a set of Indirect Safety Culture Indicators and …".
Please elaborate on how and on what bases these Indirect Safety Culture Indicators were developed, and provide detailed information on the elements that constitute such indicators.

1) As was stated in the National Report, article 10, the development of a set of indirect safety culture indicators are in progress. The first approach to this subject was to select from the current safety indicators list those indicators that indirectly are in connection with safety culture attitudes. In this selection are included Indicators related with:

Training:

- Number of hours devoted to training on safety-related issues.

Feedback from Operational Experience:

- Number of documented event analyses, findings or design modifications in similar power plants.

Internal Control:

- Number of internal technical audits.

Compliance with Regulatory Authority standards

- Number of pending Regulatory Requirements.
- Number of violations to the Mandatory Documents.

Abnormal Operation

- Number of relevant events. When direct or root causes are associated with human deficiencies
- Safety Systems actuation's. Considering here only the ones related with human failures

With the advance of the project possibly more indirect indicators would be included, such as indirect indicator related with human performance indicators, regulatory audits findings and resident inspectors findings.

In addition, recognizing the complex subject, Argentina has a close follow up on the international development in this area development in this area

N° 33
CNS-Ref.-Art.: 10
Page of Report: 31
Chapter of Nat. Report: 10

In the last paragraph of the Article 10 some other activities were discussed, which are "in progress such as development of a set of Indirect Safety Culture Indicators and improvement of regulators training to deal with "organisational factors" (which are imperative to safe performance in Nuclear Power Plants)".
What kind of indicators for the safety culture evaluation are under development?. Please, provide the short definition and purpose of these indicators.

1) As was stated in the National Report, article 10, the development of a set of indirect safety culture indicators are in progress. The first approach to this subject was to select from the current safety indicators list those indicators that indirectly are in connection with safety culture attitudes. In this selection are included Indicators related with:

Training*:*

- Number of hours devoted to training on safety-related issues.

Feedback from Operational Experience:

- Number of documented event analyses, findings or design modifications in similar power plants.

Internal Control:

- Number of internal technical audits.

Compliance with Regulatory Authority standards

- Number of pending Regulatory Requirements.
- Number of violations to the Mandatory Documents.

Abnormal Operation

- Number of relevant events. When direct or root causes are associated with human deficiencies
- Safety Systems actuation's. Considering here only the ones related with human failures

With the advance of the project possibly more indirect indicators would be included, such as indirect indicator related with human performance indicators, regulatory audits findings and resident inspectors findings.

In addition, recognizing the complex subject, Argentina has a close follow up on the international development in this area.

N° 34
CNS-Ref.-Art.: 10
Page of Report: 31
Chapter of Nat. Report: 10

Which definition of Safety Culture is used in Argentina?.
Which criteria are applied to determine "declining safety culture"?.
How are new developments and concepts in the area of safety culture considered (e.g. IAEA SCEPT, SCART, TECDOC 1329, etc.)?

Part a): Not only one definition of Safety Culture is used. The Regulatory body and the Responsible Organization are in agreement to use the definition stated on point 3.4 TECDOC 1329. In addition, the three-stage model is continuously applied. At present, evaluations shown that most of the regulated activities (NPP´s) have achieved the 3rd stage (continuous improving)

Part b): To determine “declining Safety Culture” the Regulatory Body assess the results using all the regulatory tools such as inspections, audits, Operating Experience Feedback, Program results and Safety Performance Indicators results.

On the other hand, the Responsible Organization has been used the IAEA draft on Safety Practice “Developing a Safety Culture. Practical suggestions to assist Progress” to prepare queering and interviews among the NPP personnel to determine their perception on Safety Culture aspects and to define the issues that require more emphasis, such as communications, working as a team, conflicts resolution, problems focusing, third parties relationship, openness, production-safety relationship, long/short term views, high level/low level staff relationship. The result of such queering and interviews led to realize detecting opportunities of improvement.

Part c): New developments such as those detailed in the IAEA TECDOC 1329 are in permanent consideration. Safety Culture Enhancement Program services and SCART missions are not been considered until now.

N° 35
CNS-Ref.-Art.: 10
Page of Report: 31
Chapter of Nat. Report: 10

During the second review meeting Argentina reported about ambitious activities with regard to safety culture. Have the chosen approaches and practices proven to be effective or have there been changes?

The approaches and practices applied were effective, however, safety culture tools are in continuous improvement. The relevant areas where the safety culture practices have been applied were:

- Regulator/Operator relationship, and
- Operator attitudes to deal with minor events.

N° 36
CNS-Ref.-Art.: 10
Page of Report: 31
Chapter of Nat. Report: 10

The report describes additional activities, which were carried out to improve the promotion and the evaluation of safety culture, without specifying which action came from the regulator and which came from the operator. In addition the reports does not shown any indication about the trending of the safety culture awareness. Could Argentina illustrate by facts the impact of these measures on safety (for instance the number of event of near misses with the time, or some other indicator)?

The additional activities mentioned in the National Report carried out to improve and evaluate safety culture come from both regulator and operator.

Follow up of training programs completed by operators showed an increase of safety culture in many of the training courses. Example of these activities is the training in “lessons learned” from Operating Experience Feedback (National and International). During the past 3 years approximately 15 Significant events were reported by NPPs. From those events about 5 root / contributed causes related with human factors have been found. None of them were the main cause of the occurrence of the event. In addition, human related contribution on minor events and near misses has been decreased near 40% since the last Nuclear Safety Report. It can be seen as an improvement in safety culture.

Relationship regulator/operator have shown an important advance in safety culture. As an example, all the major findings from regulatory audits carried out since last years are good received by operators and corrective actions are implemented as soon as practicable. Additionally, 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 last years.

N° 37
CNS-Ref.-Art.: 11
Page of Report: 34
Chapter of Nat. Report: 11.2

The report addresses funds for decommissioning in the answer to question No. 25 in Annex III. Please provide some information on the current situation regarding the assignment of financial guarantees/provisions for decommissioning, particularly after the complications caused by the economic difficulties of the years 2001-2003.

The National Decree 1390/98 states the establishment of a Radioactive Waste Management Fund to meet the decommissioning expenses of each NPP, with contributions by NASA of approximately 2% of their income from the nuclear power generation. Such Decree also establishes that the financial contributions to the Fund will begin after privatization of NPPs. Considering that privatization was not carried out, CNEA as Responsible Organization for the decommissioning process, is analyzing different alternatives to carry out the decommissioning of the NPPs. On the other hand, CNEA and NASA are financing the management activities with their own resources until the Fund be constituted.

N° 38
CNS-Ref.-Art.: 11
Page of Report: 36
Chapter of Nat. Report: 11.1

The report sets out the 2001, 2002 and 2003 annual expenses of NASA. By which financial resources is this company funded? How were safety improvements financed during the reporting period? Which provisions have been made regarding decommissioning and radioactive waste management at the nuclear installations?

Part a) and b): The financial resources of NASA are funded mainly from its own resources (selling electricity). Such resources were applied to current expenses and safety improvements for both NPPs.

Part c): The National Decree 1390/98 states the establishment of a Radioactive Waste Management Fund to meet the decommissioning expenses of each NPP, with contributions by NASA of approximately 2% of their income from the nuclear power generation. Such Decree also establishes that the contributions to the Fund will begin after privatization of NPPs. Considering both the privatization delay and the lack of definitions of such process, CNEA as Responsible Organization for the decommissioning process, is analyzing different alternatives to carry out the decommissioning of the NPPs. On the other hand, CNEA and NASA are financing the management activities with their own resources until the Fund be constituted.

N° 39
CNS-Ref.-Art.: 11
Page of Report: 34
Chapter of Nat. Report: 11.2

The report states that "qualification programs and retraining of the operation personnel …" continued. Please explain how the simulator-based training programs were used for the said purpose.

The Brazilian simulator belonging to Electronuclear enterprise is used annually to train the Atucha I NPP personnel. The simulator was constructed according to Angra-II NPP features, but some program modifications allow to use it for Atucha I NPP personnel training, in particular the main backfitting aspects were also included in the simulator. Besides, at Atucha I Training Department there is a dedicated Interactive Graphic Simulator where the plant personnel is trained.

The Embalse NPP personnel is trained in the full scope simulator of the Gentilly II NPP at Canada which was adapted for this purpose.

N° 40
CNS-REF.-ART.: 12
PAGE OF REPORT: 39
CHAPTER OF NAT. REPORT: 12

The report only very briefly mentions the "methods to prevent, detect and correct the events related with human factors". Would it be possible to obtain some data related to those events in the past three years, which would allow the reader to make its own opinion about the efficiency of these methods and possible lessons to be learnt by the other Contracting Parties?

Within the Operating Experience Feedback Program, methods commonly used such as Human Performance Investigation Program (HPIP) and Human Performance Enhancement System (HPES) and their associated techniques to analyze root and direct causes have been applied. Regulatory Body staff participated in a IAEA Coordinated Research Program (CRP) about the efficiency of such methods used to evaluate events. The conclusions of the CRP are shown in the IAEA-TECDOC-1278 "Review of methodologies for analysis of safety incidents at NPPs" - Final report of the research program.

During the past 3 years 15 events were reported by NPPs. From those events about 5 root causes related with human factors have been found. None of them were the main cause of the occurrence of the event. Whenever was 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.

Some examples where complete analysis (included lessons learned) are as follow:

- CNA-I: Manual Shutdown to repair mantel in N-30 fuel channel.
- CNE: Manual Shutdown after opening of primary system liquid relief valve.
- CNE: Refuelling Machine heavy water spilling .

N° 41
CNS-REF.-ART.: 12
PAGE OF REPORT: 39
CHAPTER OF NAT. REPORT: 12

What progress has been achieved since 2002? Are there any regulatory requirements for the licensees to set up safety management systems? Have the licensees already installed safety management systems?

Part a): In connection with Article 12, the progress achieved can be summarized as follow:

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

Human reliability improvements in Atucha I and Embalse NPP reviewing normal operating procedures and emergency procedures taken into account in the PSAs results.

Fostering the training program addressing past wrong human behavior and mistakes.

Widespread personnel re-training using international and national events

Part b): There are no specific regulatory requirements related to safety management systems.

Part c): The management system as defined on Draft IAEA Safety Standard Series DS338 is not implemented. However, within the quality system of NPPs there are many principles and objectives considered in similar sense.

N° 42
CNS-REF.-ART.: 12
PAGE OF REPORT: 39
CHAPTER OF NAT. REPORT: 12

It was stated that "…PSAs carried out for Atucha 1 and Embalse (have) shown that human corrective actions were necessary to be considered in order to ensure that the capabilities and limitations of human performance were taken into account in the procedures for normal and abnormal operation".

Please provide detailed information on this subject matter and give specific examples of the PSA results that highlighted this issue.

Human corrective actions carried out at the plants as a result of PSAs were the following:

Embalse NPP:

- Emergency water supply system
  1. Emergency Procedure modification to improve the operator reliability to provide lake water supply in case of refill dousing water.
  2. System test procedure modifications.
  3. Operators actions modification due to procedures mistakes.
- Emergency Core Cooling System
  4. Emergency Procedure modification for recovery actions of specific components.
  5. System test procedure modifications to avoid misalignments.
  6. Some components were included within the surveillance program.
  7. Operator actions Improvement related to high/low pressure manual connection.
  8. Changes from manual to automatic actions of specific operator actions to improve reliability.
  9. Operators actions modification due to procedures mistakes.
- Service Water System
  10. Emergency Procedure modification for recovery actions to supply electric power in case of loss of service water.
  11. Better procedures to improve the operator reliability to consider the Emergency water supply system en case of loss of service water.
- Feedwater system
  12. Links improvements among emergency operating procedures
- Moderator system
  13. Shutdown test improvements
- Electrical system
  14. Maintenance improvements for specific components
  15. Test improvements

Atucha I NPP:

Many human actions improvements were described in the previous reports. The backfitting program included a large number of design changes and procedures modifications. After the backfitting 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 were carried out modifying procedures and increasing training efforts. Additionally, new systems were included and as a consequence new procedures were carried out improving the overall plant safety. The size of the backfitting program implied the review of all plant procedures and the critical human actions that combined with the PSA results give rise to improve its reliability.

Last procedure changes were the following:

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

N° 43
CNS-Ref.-Art.: 12
Page of Report: 39
Chapter of Nat. Report: 12

Australia notes that that report refers to ongoing activity in this area since the previous National Reports. What outcomes have been identified from the different methodologies used by the Responsible Organisation and by the Regulatory Body? What corrective actions have been taken as a result

The following results and corrective actions were identified:

- Procedures: lack of completeness, lack of clear acceptance criteria, mistakes and poor human action reliability. Corrective action: Changes and update in operating procedures.
- Training: lack of specific training, inadequate training. Corrective action: changes and update in personnel training and retraining.
- Plant Systems: inadequate design and poor ergonomic design. Corrective action: upgrading of systems and components.
- Additionally, other results of applying different methodologies were found such as: inadequate communication, lack of planning, lack of supervision and lack of resources. Corrective action: Changes in the corresponding management policies.

N° 44
CNS-Ref.-Art.: 12
Page of Report: 39
Chapter of Nat. Report: 12

A comprehensive illustration – even if it is copied from former reports – would help to better analyze the Argentinean human factors activities

This comment posted as a question, will be taken into account in the next national reports. However, the previous national reports include a detailed information about the human factors activities that can be reviewed also from the web-site.

N° 45
CNS-Ref.-Art.: 13
Page of Report: 101
Chapter of Nat. Report: Annex III (q 37)

Question no. 37 in Annex III refers to Figure 13.1 in Article 13 and it states that the Regulatory Body is empowered to review the quality programs of both the Responsible Organisation and the nuclear installation, and the quality system and programs must meet the regulatory standard 3.6.1 "Quality System". In the review process how does the Regulatory Body treat quality management systems that deviate from requirements of the regulatory standard AR 3.6.1 (e.g. based on ISO 9001:2000, 10CFR50 App. B, national standards, etc.)?.

The management system as defined on Draft IAEA Safety Standard Series DS338 is not implemented. However, within the quality system of NPPs there are many principles and objectives considered in similar sense.

In case of non-compliance with regulatory standard AR 3.6.1; according with regulatory procedures, it is considered a critical non-compliance (non-conformity). The regulatory measures taken involve immediate correction of the non-conformity.

N° 46
CNS-Ref.-Art.: 13
Page of Report: 41
Chapter of Nat. Report: 13

The General Quality Assurance Manual defines the frame for the Quality Management Systems of the individual NPPs. A short description of the contents of the different parts of this general manual would be preferable, together with the competences for the variations in the manuals in the various NPPs.

The article 13 "Quality Assurance" have been presented in a reduced form and only were highlighted the main QA Program contents and the requirements that have being fulfilled.

Argentina are agree with in considering that a brief content description of the different parts of the General Quality Assurance Manual will be useful to a more comprehensive understanding.

Therefore, Argentina will consider this suggestion and, in case it were accepted, it could be possible to extend the above mentioned description in the next national reports.

However, if Switzerland have some particular interest about this issue, Argentina could send a General Quality Assurance Manual copy (Spanish version).

N° 47
CNS-Ref.-Art.: 14
Page of Report: 45
Chapter of Nat. Report: 14

This paragraph indicates that "in this report the main safety assessment efforts were explained in detail within Article 6". Furthermore, in Chapter 7.3 it is stated that "it is a regulatory requirement to perform and update the NPPs probabilistic safety analysis….".
What is the scope of the risk analyses in terms of PSA levels as well as scope of initiating events and operational modes?. How frequently are the risk analyses updated?

The present status of Probabilistic Safety Analysis (PSA) for the plants in operation including scope and revisions is shown below.

PSA is considered a living document and it is included as a part of the Mandatory Documentation for a plant to operate. PSA must be updated to reflect the actual plant safety. There are two ways required to update the PSAs, 1) every plant design change or procedures updating that can imply any safety impact and 2) when performing the PSR.

As an example, PSA updating was one of the regulatory requirements for restarting the plant when Atucha I backfitting was implemented.

Atucha I NPP
PSA Level 1.

- *Source of radioactive release* : Reactor core
  Operational state Nominal full power operation
  Initiating Events Internal events
  - First quantification Rev 0 completed in 1996
  - First revision including plant modifications resulting from PSA recommendations was completed in 1998.
  - Second revision including complete plant backfitting was completed in 2003.
- *Source of radioactive release* : Reactor core
  Operational state Nominal full power operation
  Initiating Events Internal fires
  - Completed in 2000.
- *Source of radioactive release* : Sources different from reactor core
  - Completed in 2000.
- *Source of radioactive release* : Reactor core
  Operational state cold depressurized shutdown state
  - Completed in 2001.

Embalse NPP
PSA Level 1**.**

- *Source of radioactive release* : Reactor core
  Operational state Nominal full power operation
  Initiating Events Internal events
  - First quantification completed in 2001
  - First revision completed in 2003.
- *Source of radioactive release* : Reactor core
  Operational state Nominal full power operation
  Initiating Events Internal fires
  - In progress (deadline: second quarter 2005).
- *Source of radioactive release* : Sources different from reactor core
  - Completed in 2003.
- *Source of radioactive release* : Reactor core
  Operational state Reduced power and shutdown states
  - In progress (deadline: second quarter 2005).

Atucha II NPP
A preliminary level 3 PSA for Atucha II was carried out.

N° 48
CNS-Ref.-Art.: 14
Page of Report: 47
Chapter of Nat. Report: 14.2

Please give more details on the experience with safety and performance indicators. What is the justification for the values of acceptability criteria for these indicators? Has this proven as an effective regulatory tool to monitor the operator's safety performance?

Part a): As result of the pilot implementation experience, it was defined an acceptability range for each indicator instead of a threshold values. Such acceptability range or satisfactory zone is considered as a range of normal behavior of the indicator. The main reason was indicators frequency distribution does not allow establishing representative thresholds.

The preliminary criteria selected to define satisfactory zone boundaries are the following:

- The boundaries of the satisfactory zones are the most probable range of the *frequencies* distribution. These values are acceptable from regulatory point of view.
- If an indicator has had periods with different frequency distributions, the most conservative distribution *was adopted*.
- If an indicator is strongly dependent on operational conditions, the actual condition *was adopted*.
- If indicator statistics are not satisfactory from regulatory point of view, boundaries are defined according to regulatory expectations.
- Boundaries could be modified according to indicators evolution when trends indicate *a better performance.*

Three aspects are considered to evaluate the indicators:

1. Comparison of the indicator value with respect to the satisfactory zone
2. Trends of the indicator over the last *year*.
3. Additional information related for particular indicator *(examples: increase of dose during outages).*

Part b): The set of safety performance indicators is used as a regulatory tool to provide an additional view of the nuclear power plants performance allowing to improve the ability to detect any eventual degradation on safety related areas. It is a satisfactory tool but not using it stand-alone, but together with another tools (such as event analysis, audits, PI, (among others) for monitoring safety).

N° 49
CNS-Ref.-Art.: 14
Page of Report: 46
Chapter of Nat. Report: 14.1.2.3

**This paragraph mentions that "…it has been detected flow assisted corrosion (FAC) …".
Since some other reactor units in other countries may have experienced FAC, please indicate how Argentina shares and benefits from sharing OPEX and lesson learned with other countries.**

The Operating Experience Feedback program applied by the Responsible Organization includes to share the information and lesson learnt from CANDU Owners Group (COG), WANO, IAEA-IRS, IAEA-INES, specific meetings and workshops.

The Regulatory Body share the information with IAEA-IRS, CANDU Senior Regulators Group, Bilateral Regulatory agreements, Ibero-American Forum of Regulators, IAEA-INES, specific meetings and workshops.

In connection with FAC applied at Argentinean NPPs, the event happened at Mihama 3, Canadian operating experience and WANO and INPO reports, the information was disseminated and it was included in the training and inspection programs. Additional inspections hold points regarding FAC were included in the shutdown inspection programs.

N° 50
CNS-Ref.-Art.: 14
Page of Report: 46
Chapter of Nat. Report: 14.1.2.1

The report indicates that "… 253 pressure tubes … have been repositioned…". Please explain whether repositioning of pressure tubes means repositioning of the garter springs or it is physically repositioning the pressure tube.

In that paragraph the expression "repositioning of pressure tubes" means repositioning of garter springs.

N° 51
CNS-Ref.-Art.: 15
Page of Report: 57
Chapter of Nat. Report: 15.3.1.2

The training of the workers is partially described in the section on ALARA activities. However, this centres only on the personnel performing certain tasks and there is no indication of whether they receive specific training on Radiological Protection.

How are the radiological protection training requirements for the plant professionally exposed personnel and for off-site or contracted workers defined and established?

The Basic Safety Standard, AR 10.1.1, establishes that all plant personnel related to radiation protection field must receive an adequate training according with each specific position. In this sense, the NPPs have elaborated specific procedures to comply with. The following three levels of training have been defined:

- Licensed personnel belong to the plant organization: individual license and specific authorization is required. Specific authorization requires retraining for renewal according to the training annual program.
- Non-licensed personnel belong to the plant organization: basic training course on radiation protection and retraining according the training annual program.
- Contracted personnel to perform specific task that does not belong to the plant organization: basic and specific training course on radiation protection.

N° 52
CNS-Ref.-Art.: 15
Page of Report: 49
Chapter of Nat. Report: 15.1

Missing legend: Are these annual discharge limits? From which dose limit are these limits deduced? Where is the model determined (reference)?

Part a) and b): Table 15.1 includes Ki values that limit the environmental discharges applying the following expression:

Where:

Ai is nuclide i activity released to the environment in the period considered
Ki is a constant activity value, stipulated for the nuclide i, for a given installation
L =1 in a year.

Each Ki corresponds to a critical group annual individual dose of 0.05 mSv. The value of Ki is calculated for each installation, radionuclide and type of discharge (liquid and gaseous) using specific models to estimate the dose to the critical group, taking into account the site characteristics and the critical group location. This kind of evaluation ensures that if this expression is satisfied, the dose constraint for public will be not exceeded.

Part c): The models used are known as "concentration factor methods" (ref. IAEA Safety Series Nº 57 " Generic Models and Parameters for Assessing the Environmental Transfer of Radionuclides from Routine Releases" (1982) and IAEA Safety Reports Series Nº 19 "Generic Models for Use in Assessing the Impact of Discharges of Radioactive Substances to the Environment" (2001). Such models contain specific plant site information: critical group location, habits, food consumption and local dispersion factors.

N° 53
CNS-Ref.-Art.: 15
Page of Report: 49
Chapter of Nat. Report: 15

Is there any provision at the regulatory body for cross checking the results of gaseous and liquid effluents discharged into the environment say through an independent laboratory measurement?

Each NPP measure and inform the release activities as required. Regulatory Body has an audit program applied to the release measurement procedures and it carries out benchmarking exercises on pattern measurements, procedures control, measurement devices and calibration.

Besides, the Responsible Organization and the Regulatory Body performs independent measurements of activity concentration on environmental samples and public dose evaluation using environmental models that considers "concentration factor methods" as recommended in the IAEA series 19 and 57. Each model has specific plant information such as critical group location, habits and food consumption and dispersion local factors of environmental releases.

N° 54
CNS-Ref.-Art.: 15
Page of Report: 52, 55
Chapter of Nat. Report: 15.1, 15.2

It is reported that "The average normalized effective collective dose for the period 2001-2003, due to C-14 releases is a bit higher than the collective effective dose constraint per unit of electric energy generated established in Standard 3.1.2 due to the fact that Embalse´s design was finished before the above mentioned standard was in force".
Please explain what plans will be established to upgrade the station to reduce the effective dose to limits set out in Standard 3.1.2.

The regulatory Standard AR 3.1.2 "Radioactive Effluents limitation in NPPs" establishes requirements for design of NPPs. Such standard is not applicable for Embalse NPP due to its design was finished before the above mentioned standard was in force.

In spite of the small difference regarding the discharge limit established for 14C in Embalse NPP (18 manSv /GWy, in comparison to 15 manSv /Gwy Standard AR 3.1.2) there is no justification to implement a retention system for 14C to upgrade the station.

N° 55
CNS-Ref.-Art.: 15
Page of Report: 49
Chapter of Nat. Report: 15

a) In this third report, the general information is missing. The report references the former two reports on CNS, wherein the most information is included.

Argentina has imposed an "Alpha-Value" of 10'000 USD/man-Sv, which is very low in comparison to the alpha-value used for protection of NPP personal in the US, Germany or Switzerland. How have you determined this value?.

b) In all three reports the following information is missing:

- **Dose limit for personal under exposure not related with their work.**
- **Dose limits in emergency conditions.**
- **Dose limits during chronic exposure.**

c) Release of low level radioactive, "solid" material (clearance of material originating from areas under authority control).

d) Nothing is written about how ALARA is ensured on the occupational exposure during normal operation and outages in NPP.

e) Environmental radiological surveillance: Values are reported but information is missing on how the surveillance has been performed/realized.

f) Regulatory control activities especially in the radiation protection field.

Please give the missing information.

Part a): The value of 10,000 USD/manSv is used in the differential cost-profit analysis for the optimization of radiological protection systems. It should be added that the systems of radioactive effluents retention of nuclear installations must be optimized according to the corresponding regulatory standard.

The above mentioned value was established taking into account the international trends during the seventies.

In fact, as may be noticed in the following references, the monetary value of the ratio of social cost to collective dose fluctuated between 1000 USD/manSv and 25000 USD/manSv.

Reference	USD/Sv-man
ICRP Publication 22 [1]	1 000 – 25 000
Application in the Nordic Countries of ICRP Pub. 26 [2]	1 000 – 20 000
Limitation of Releases of Radioactive Substances from Nuclear Power Stations [3]	1 000 – 25 000

(1) 1973.
(2) The Radiation Protection Institutes in Denmark, Finland, Iceland, Norway and Sweden, 1976.
(3) The Swedish National Institute of Radiation Protection, 1977.

However, the Regulatory Body has considered the review such value in the near future together with the update of AR 10.1.1 Basic Safety Standard and the new ICRP recommendations.

Part b)

- Dose limit for personal under exposure not related with their work: In this case it is applied the public dose limit of 1 mSv per year .
- Dose limits in emergency conditions and dose limits during chronic exposure:
  The Regulatory Standard AR 10.1.1. "Radiological Safety. Basic Standard" establishes:
- Radiological intervention is justified if it foresee that such intervention lead to an improvement of the situation.
- The intervention is generally justified when the estimated dose could exceed the severe deterministic threshold.
- The form, scale, and duration of the intervention shall be optimized. Dose limits established for practices are not applicable to such interventions.

Additionally, regarding the exposure of individuals that carry out the interventions the Regulatory Standard establishes:

- When the estimated dose exceed 100 mSv, the task to be performed during an intervention shall be on a voluntary basis. Volunteers shall be previously informed about the risk.
- Intervention situations which imply volunteers effective dose that exceed 1 Sv or equivalent skin dose higher than 10 Sv, shall only be justified to save human lives.

Part c): Release of low level radioactive, "solid" material (clearance of material originating from areas under authority control)

The criteria applied in Argentina to the release of low level radioactive solid material have been presented in the National Report to the Joint Convention on the Safety of Spent Nuclear Fuel and on the Safety of Radioactive Waste Management, Revision 1, 2003.

For clarification the above mentioned criteria is transcript in the following paragraphs.

The Section A of the National Report to the Joint Convention states:

"…the definition of radioactive waste has been specified understanding that it includes:

- **radioactive disposable waste**, means materials that on account of their concentration of radioactivity and/or total radioactivity, cannot be dispersed into the environment and therefore, require treatment, conditioning and final disposal.
- **dischargeable waste**, means the planned and controlled discharge into the environment of liquid and gaseous radioactive materials that originate from the normal operation of a nuclear facility on account of their total radioactivity.

- **clearable (clearance) radioactive materials**, means radioactive materials that on account of their concentration of radioactivity and/or total radioactivity may be released from regulatory control after a limited storage period for decaying."

The Section B of the above mentioned report states:

"…radioactive waste subject to clearance from regulatory requirement and dischargeable waste, it is the regulatory framework and the regulations in force that define the criteria for their management. In all cases, safety methods applied have been the result of discussions and international consensus on this matter."

The Section B.4 of the mentioned report gives the following details on the matter:

"In Argentina the following criteria is applied to Radioactive Waste Management:

a. Allow the decaying of radioactive materials which on account of their concentration of radioactivity and/or total radioactivity may be released from regulatory control after a limited storage period compatible with their safety.
b. Authorize the planned and controlled discharge of liquid and gaseous radioactive materials that originate from the normal operation of a nuclear facility and which on account of their total radioactivity may be released into the environment.
c. Treatment, conditioning and final disposal of radioactive waste, understanding that radioactive waste means materials that on account of their concentration of radioactivity and/or total radioactivity cannot be released into the environment.

In the first instance the Regulatory Authority sets forth acceptable doses for the case of release of material from regulatory control in accordance to the exemption criteria. As provided in AR 10.1.1. Radiological Safety Standard, the effective doses constraint value for exemption is 10 µSv/year for individuals most exposed to radiation and 1Sv man/year as an effective collective dose value."

Further, each NPP has internal procedures to segregate the wastes according to its activity as disposable waste and clearable radioactive material . The management of the later includes a screening procedure, the activity measurement of the material, its registration and further release.

Part d): ALARA program is carried out in both NPPs during normal operation and during outages. Each NPP have 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.

In the National Report is shown some of the main improvements of the plant with high doses activities during outages applying ALARA to perform: control rod tube guide replacement, steam generators inspection, main coolant pumps seal replacement and channel replacement.

Part e): Each NPP measures and reports the radioactive releases as required. Regulatory Body has an audit program applied to the release measurement procedures and it carries out benchmarking exercises on pattern measurements, procedures control, measurement devices and calibration. Besides, the Responsible Organization and the Regulatory Body perform independent measurements of activity concentration on environmental samples and public dose evaluation using environmental models that considers "concentration factor methods" as recommended in the IAEA Series 19 and 57. Each model has specific plant information such as critical group location, habits and food consumption and dispersion local factors of environmental releases.

Part f): The control and surveillance of the compliance with standards and other regulatory documents, are completed with a program of routine and non routine inspections and audits, which contribute to determine the fulfilment of the operating license and every other mandatory documentation.

Resident inspectors carry out this control and different working groups belonging to the Regulatory Body, who perform analyses and evaluations related to different topics on Radiological Safety. These working teams have their own laboratories so that they are able to perform the measurements and experiments required for such purpose.

These controls are performed routinely, but they are especially carried out when it is necessary to reinforce the inspection tasks, as in case of programmed outages and non foreseen shutdowns, or as a consequence of some specific situation.

The periodic test program related to radiological protection is monitored and observed during its performance. Experimental data coming out from the mentioned tests are confirmed. Among this set of tests, those related to radiation detection equipment installed in different working areas, and execution of the emergency plan implementation exercises are outstanding (see Article 16).

The personnel dosimetry system is evaluated not only for external irradiation but also for internal contamination, by means of specific audits carried out by Regulatory Body specialists, requiring the participation of dosimetry laboratories in intercomparison exercises.

These exercises are annually performed by the Regulatory Body through the use of its own laboratories together with the support of the Secondary Laboratory of Dosimetry Calibrations (National Atomic Energy Commission).

Concerning the control of effluents released to the environment by the installations, the present measurement plan during a year operation consists in the measurement of the released activity in those places where effluents are emitted. This plan includes a routine measuring timetable and it is complemented with controls at random.

In addition to the environment monitoring plan carried out by the installations, the Regulatory Body independently performs environmental measurements in the surroundings of Atucha I and Embalse nuclear power plants or nearby zones with its own laboratories and specialists. The set of control points, where samples for this plan are taken, includes not only those selected by the installation but also some other points chosen according to the Regulatory Body criterion.

N° 56
CNS-Ref.-Art.: 15
Page of Report: 51-55
Chapter of Nat. Report: 15.1, 15.2

Please provide information as to whether analyses have been completed and trends have been noticed over time for exposure dose rates. Please also indicate whether any conclusions have been reached on reasons/causes for any trends

The trends on the public exposure doses which are directly related with the discharges were analyzed.

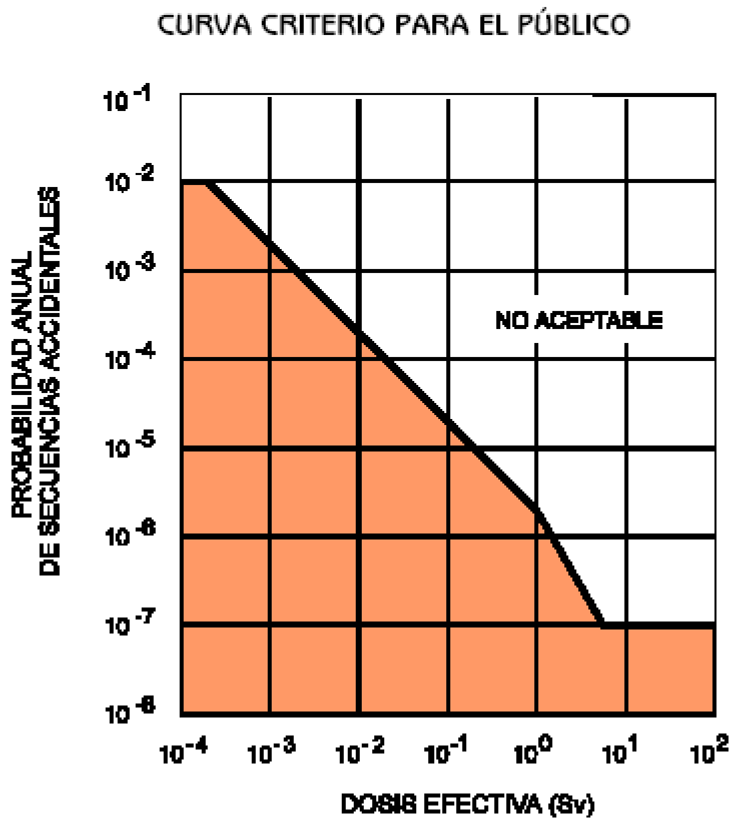
In the Atucha I NPP particular case, the critical group doses increasing trend corresponding to 2001 / 2003 period, was due to an increment of the gaseous discharges. The reason were both the backfitting tasks carried out during 2002 and some deficiencies in the fuel channel seals during 2003. The above mentioned reasons caused both tritium and noble gases discharges increase.

N° 57
CNS-Ref.-Art.: 15
Page of Report: 49
Chapter of Nat. Report: 15

Which acceptance criteria have been used for the regulatory review of the radiological consequences of design basis accidents? Are these criteria related to releases or related to radiological exposures? If dose limits are applied, which are the parameters (e.g. exposure pathways, integration times, distances) considered for the calculation?

Part a) and b): As a part of the licensing process, to grant the Construction or Operating Licenses the regulatory Standard AR 3.1.3 “Radiological Criteria related to NPP Accidents” must be met. The range of accidents covered for this standard are not classified discriminating design basis and beyond design basis accidents but considering that any accident can potentially occur and they has two parameters associated, its occurrence probability and its consequence. The consequences are measured in terms of individual doses.

The Regulatory Body requires that the Safety Reports must be updated according to the state of the art at the licensing time, if applicable. However, in all cases, based on AR 3.1.3 the overall plant risk must be acceptable.



Part c): Individual doses used in the acceptance criteria are the maximum individual doses corresponding to each accident, with its associated probability, assuming that the reference person considered belongs to the worst case in terms of received dose without the application of any early emergency accident countermeasures. Meteorological aspects are also considered, in particular the consideration of atmospheric stability classes (PASQUILL) and wind direction, speed and frequency.

Dose limits are not applied in such standard although risk limitation criteria are in place. The assessment the accidental doses the following parameters are used:

- Exposure Pathways: cloud immersion external dose, ground and surface deposition external dose, inhalation dose, ingestion dose.
- Integration time: 50 years.
- Distance: worst case in terms of received dose.

N° 58
CNS-Ref.-Art.: 15
Page of Report: 55
Chapter of Nat. Report: 15.3

Licensee is responsible for providing personnel monitoring facility and assessment of doses to the workers. Do the dosimetry laboratories need a formal approval of the regulatory body. Are regulatory standards defined for establishment of dosimetry laboratories and for providing dosimetry services?

Concerning the Regulatory Body, there is not a formal approval of dosimetry laboratories and there is no specific standard for establishment of dosimetry laboratories and for providing dosimetry services. However, the laboratories must comply formal National IRAM Standards.

The Regulatory Body controls the quality of the data reported by the Responsible Organization verifying the measurement systems, procedures and calibration equipments. Additionally, benchmarking exercises in personnel dosimetry activities are carried out by the Regulatory Body.

N° 59
CNS-Ref.-Art.: 16
Page of Report: 59 to 62
Chapter of Nat. Report: 16

The report does not explicitly identify any activity relating to site emergency plan drills or simulations.
In addition to covering initial and on-going training on the response outside nuclear power plants, do the nuclear emergency exercises performed also cover practices at the plant itself to verify the site emergency plan?

The Emergency Plans for both NPPs in operation (Atucha I NPP and Embalse NPP) have dedicated chapters that considers off-site and on-site aspects. The off-site aspects are under the Regulatory Body responsibility, regarding the public radiation protection and the involved Emergency Organizations. The Responsible Organization acts on behalf of the Regulatory Body during the first hours from the beginning of the accident until the emergency control is taken by the Regulatory Body and during this time it is responsible for applying the first protection measures. The on-site aspects are always under the Responsible Organization responsibility.

According to the Operation License, an annual emergency exercise is carried out. On-site emergency exercises are performed to test the capability of all plant emergency organization to mitigate the accident and to minimize its radiological consequences. Additionally, abnormal event procedures are trained for Atucha I and Embalse NPPs at the full scope simulators.

In the on-site exercises, postulated event sequences that lead to core damage and radionuclide releases are defined. During the accident progression different plant response groups take part. Initially, reactor operation team follow the emergency procedures and it requires the action of plant response groups and reports to external organizations. Later, several actions like sheltering, evacuation, and also thyroid prophylaxis of plant workers (simulating the use of potassium iodide -IK- pills) are implemented. A radiological control of plant staff is performed during the exercise.

Once the emergency exercise is concluded wrap-ups and discussions to consider the different findings are carried out and, a report including strengths and weakness is issued. Finally, corrective actions are taken, if necessary.

The Regulatory Body carry out a close follow-up of the different emergency exercise stages producing findings independently the Responsible Organization.

N° 60
CNS-Ref.-Art.: 16
Page of Report: 61
Chapter of Nat. Report: 16.2

The inhabitants living in the surroundings of the NPPs participate in emergency exercises (e.g. in Lima, Atucha I NPP 2002/2003 up to 6000 inhabitants). Which experiences were made with the participation of the inhabitants?

The inhabitants living in the surroundings of the NPPs receive information and training. Written information is provided and they are invited to participle on a voluntary basis in lectures about the exercise.

During the emergency exercise, the inhabitants practices all the protection measures learnt with the participation of all involved organizations and competent authorities. All the inhabitants receive (at home, shops, etc.) brochures containing information related to the exercise and candies that simulates be IK (potassium iodide) pills. Also, it is recommended sheltering preparation and control access (by cutting off routes and navigable ways informing about the emergency exercise).

Communication to inhabitants is performed through the means available in case of emergency which are messages by local FM radio (broadcasting) and using sirens and loudspeakers.

As general experience many inhabitants participate of the exercises following instructions, it means, remaining inside the buildings (sheltering), listening the local FM radio following indications. Exercises are well received by the population. The Municipal Civil Defense, the Responsible Organization and the Regulatory Body are responsible for the organization of emergency exercise related to inhabitants participation.

N° 61
CNS-REF.-ART.: 16
PAGE OF REPORT: 59
CHAPTER OF NAT. REPORT: 16.1

The Reg. Body is responsible for the protection of the public. Are there any problems with the off-site authorities (municipalities, etc.)?

There are no any problem with the off-site authorities regarding this matter. The Regulatory Body is responsible for the public protection in those aspects related to the protection against ionizing radiation resulting from a nuclear emergency. However to comply with the Nuclear Law, arrangements for the coordination of emergency response and protocols for operational interfaces among regulatory body, operators and local, regional and national government are in force, practiced, coordinated, documented and maintained in order to allow that each organization met its responsibility.

The main Emergency Command is located at the Municipal Emergency Centre is chaired by the Regulatory Body and integrated by the Heads of organizations involved in the emergency such as Police, Health, Municipality, Firemen, etc where coordinated decisions are taken in order to apply the necessary protection measures providing logistical support and facilities.

N° 62
CNS-REF.-ART.: 16
PAGE OF REPORT: 62
CHAPTER OF NAT. REPORT: 16.3

Please explain how information on radiological risk and the nuclear emergency plans have been disseminated in Argentina in the surrounding of the NPPs?

Since more than 20 years ago, there is a direct and close interaction with the inhabitants living 10 km. surrounding the NPP. Interaction is achieved through a proactive information to public by notification, brochures and information papers distribution and presentations focused on the potential risks related with NPP accidents. Protective measures that could be applied in case of a nuclear emergency are included.

The information related with emergencies is disseminated through all inhabitants social activities, meeting forums and public places, elementary and high schools. High level of awareness on this subject is provided by the teachers and the authorities involved, therefore training activities related with this issue are obligatory. General training are organized taken into account the public characteristics like age and education level and it is encouraged the participation of both students and teachers by means of discussions and questions as the best way to disseminate the knowledge among their families allowing to transmit it to a large number of inhabitants. The public diffusion activities are performed on an annual basis near to emergency exercises to train the public in the issues related with the activities that will be performed during such exercise.

N° 63
CNS-REF.-ART.: 16
PAGE OF REPORT: 59
CHAPTER OF NAT. REPORT: 16.1

Regarding emergency exercises, which emergency scenarios were assumed in the exercises regarding the plant scenario and the related source term? Were core melt scenarios considered?

Different scenarios for nuclear emergency exercises involving core melt (that considers simultaneous failures of safety systems) are assumed. It includes as a first stage, radioactive contamination inside the containment (plant emergency response teams take mitigation actions) and a later radioactive release due to a containment failure (preparation and application of the corresponding inhabitants protective measures through external response organizations commanded by the Regulatory Body).

Postulated accidents and its corresponding source term are different from each exercise, involving radioactive releases for on-site and off-site countermeasures, in order to assess the exercise performance of the organizations included in the emergency plan.

N° 64
CNS-Ref.-Art.: 16
Page of Report: 59 – 61
Chapter of Nat. Report: 16.2

Why are access controls not established among the protective measures to be implemented in the event of emergency situations?

Access controls are clearly established as a protective measure in a nuclear emergency case. The restricted access areas are established according to plant status and meteorological condition and its location are informed to local Police Department by the Regulatory Body for roads cut off. Restricted navigable access control is carried out by Coastguards. Regulatory Body gives radiation protection support and contamination measurements.

N° 65
CNS-Ref.-Art.: 16
Page of Report: 60
Chapter of Nat. Report: 16.2

Please provide information on any conclusions/recommendations (i.e., lessons learned) that were made as a result of the emergency preparedness exercises.

Related to on-site aspects, emergency preparedness exercises allows to evaluate the performance of the staff and the equipment involves with the NPP Emergency Plan. Also, the capability to provide timely response and alerts to external organizations involved to face the emergency off-site consequences are also evaluated. Additionally to planned measures, self-assessment is carried out to detect weaknesses that could affect to equipments, procedures and personnel to face the emergency. At the end of the emergency exercises, an evaluation report is issued to take corrective actions, if necessary.

Weaknesses related to conventional communication systems, difficulties in plant sheltering for long periods, the competition between physical security and measures for the emergency as evacuation of the plant; sheltering in meeting points and the plant ingress of firefighter and others emergency response groups were found. Therefore, considering these findings specific strategies were defined to improve the practices.

Related to off-site aspects, the emergency preparedness exercises are used to evaluate the performance of the organizations involved to carry out protective measures outside the plant addressing the emergency alert system , reporting and the relevant information among such organizations.

Due to several external organizations are involved during an emergency, it is important to check the coordination capability of the Regulatory Body, according to what is established in the Nuclear Law and the Emergency Plans for off-site aspects. A communication network with all organizations involved is established to apply the whole effective protection actions for the population.

Regarding to the interaction with the inhabitants in the surrounding of NPPs, it begins several months before the emergency exercise by disseminating information, training, notification, brochures and information papers distribution and presentations focused on the potential risks related with NPP accidents. Protective measures that could be applied in case of a nuclear emergency are included.

As general experience many inhabitants participate of the exercises following instructions, it means, remaining inside the buildings (sheltering), listening the local FM radio following indications. Exercises are well received by the population. The Municipal Civil Defense, the Responsible Organization and the Regulatory Body are responsible for the organization of emergency exercise related to inhabitants participation.

The most effective way found to transmit the information are both the elementary and the high school were the training activities are obligatory for all the pupils. It is encouraged the participation of students and teachers by means of discussions and questions as the best way to disseminate the knowledge among their families allowing to transmit it to a large number of inhabitants. The public diffusion activities are performed on an annual basis near to emergency exercises to train the public in the issues related with the activities that will be performed during such exercise. The inhabitants take part of the emergency exercise by checking the application of measures such as sheltering, access control and thyroid prophylaxis. The overall interaction among the authorities, plant operators and public during the training and the participation in the emergency exercise increases the confidence, openness, and transparency in the public authorities making more effective the Emergency Plan application.

N° 66
CNS-Ref.-Art.: 16
Page of Report: 59
Chapter of Nat. Report: 16.1

The Regulatory Body is the head at Emergency Control Center, one of its responsibilities being nuclear and radiological assessment during the accident. Could ARN provide information about the tools (e.g. data acquisition system from NPPs) and equipments provided to Emergency Control Center in order to perform on-line supervision of nuclear installations status during nuclear accident?

Immediately when the emergency occurs the Emergency Control Center is established in the NPP, as established by the Operating License, managing the first actions until the Municipal Emergency Center is made up in the affected municipality. The Municipal Emergency Center manage the off-site emergency, and it is constituted by the Regulatory Body, Municipal Authorities and the head of the different external organizations as Police, Coastguard, Firefighter, etc.

In addition, a Regional Control Center is made up to give support to the Municipal Emergency Center and coordinates the national response and the international notification and assistance. The NPP and the Regional Control Center have the responsibility of supporting and give information to the Municipal Emergency Center.

Until now, Municipal Emergency Center have not a data acquisition system from NPPs and it only receives information by diverse telephone/fax, mobile and satellital telephones and VHF radio. The transmitted data are not automatic and they are re-transmitted from the NPP Emergency Control Center (plant status, meteorological conditions in the plant tower, accident conditions, etc) and from the Regional Center (meteorological data from plant alternative towers, output of codes and presentation of the information using Geographic Information Systems (GIS)).

It is planned to provide the Emergency Regional Center and the Municipal Emergency Center with an on-line data acquisition system from the NPP and radiological information from the surrounding areas.

N° 67
CNS-Ref.-Art.: 16
Page of Report: 59
Chapter of Nat. Report: 16.1

Who is responsible for the classification of emergency situations, and which classifications have been made?

When an abnormal situation arises in a NPP, the Operator is responsible to classify the initial emergency situation applying the plant emergency procedures and to notify to the external emergency organizations. On- Site and Off-Site emergency classifications are as follows:

On-Site Green Alert: when an abnormal situation is detected that compromises the nuclear safety, but no emission is foreseen. The internal emergency procedures are applied and the internal emergency organization is established. There is no correlation with an off-site alert.

On-Site Yellow Alert: Is declared by the Operator when the abnormal situation detected could leads to an emission of significant amounts of radioactive material, but there is still no emission the environment. The first radioactive material barriers could have been deteriorated, and there could be contamination inside containment building. On-Site procedures are applied and the Operator declares the Emergency.

Off-Site Green Alert. Without any delay, Operator starts implementing preventive urgent countermeasures to protect public (messages to population, IK pills distribution, preparation to control access and to sheltering). At the same time, the Operator request the formation of the external emergency organization.

On-Site Red Alarm: Is declared by the Operator when the emission of significant amounts of radioactive material has just started. Internal emergency procedures continue and the situation is notify to the external emergency organization who immediately declare the Off-Site Red Alarm and continue applying countermeasures to protect public (IK pills ingestion, sheltering, access control, evacuation, consumption restrictions, etc).

The On-site and Off-Site emergency classifications are permanently coordinated and exchange of information between both emergency response centers is guaranteed.

The emergency classification described above is for the local range (10 Km around NPP). At a National level, when Operator declare On-Site Yellow Alert he must notify the Regulatory Body. When the On/Off Site Red Alarm is declared, the Regulatory Body declare the same at a national level.

For the international notification, the Convention on Early Notification of a Nuclear Accident format is adopted, and the Regulatory Body is the National Competent Authority who declare and notify the alarms to the IAEA or the potentially affected neighbor countries.

N° 68
CNS-Ref.-Art.: 17
Page of Report: 63
Chapter of Nat. Report: 17

Which measures were taken to meet the regulatory requirements with regard to the review of design parameters corresponding to earthquakes, extreme meteorological events and man-induced events?

Earthquakes:
As was stated in the previous National Nuclear Safety Reports, after to meet the requirements about the seismic reevaluation of Embalse NPP carried out before the plant commissioning, the following additional requirements were met.

- Update and improve the program of plant response against seismic events: The plant prepared and implemented operating procedures (update and improvements) to assess actual plant physical damage and plant operational situation after the 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 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. The Operating Procedure was prepared, including the parameters for the Operating Basis Earthquake (i.e. SL-1 according to IAEA SG-50-S1 Rev.1).
- Implementation of a new digital seismic instrumentation: the plant implemented the definition, procurement, installation and commissioning of a new digital seismic instrumentation to provide data directly to plant operators for decision making and for immediate actions to check the DBE exceedence. Besides, the parameters of the cumulative absolute velocity (CAV) were defined and the operating basis earthquake (OBE) given the corresponding values for Embalse Nuclear Plant. The instrumentation installed allows to record the seismic activity providing this information directly to the operator in the control room for decision making and immediate actions checking the operating basic earthquake (OBE).
- Need to perform a seismic Probabilistic Safety Assessment (in progress), the results of the seismic PSA required by the Regulatory Body could recommend some additional changes to the plant design.

The criteria for evaluation applied was based in the National Regulatory Standards AR 3.10.1 “Protection against Earthquakes in NPPs”.

Extreme meteorological phenomena:
The re-evaluation of tornadoes and severe storms hazard performed at Atucha 1 and Embalse NPP let to confirm that the tornadoes frequency distribution do not present differences within the uncertainty margin with the corresponding distribution as it were defined at the moment that the plant began to operate.

Besides, hydrological characteristics for the Atucha I site were re-evaluated considering the statistics from recent years in the Paraná river and, as a consequence the extreme value of the flooding level was modified. Requirement related to Extreme meteorological phenomena Atucha I and Embalse NPPs:

- Reevaluation of Tornadoes and Severe Storms Hazards including transmission lines which are essential for the plant safety

The criteria for evaluation applied was based in the National Regulatory Standards AR 3.1.3 “Radiological Criteria Related to accidents in NPPs” that includes the corresponding probabilistic safety criteria

Man-induced events:
The results of man-induced events analysis carried out for Embalse NPP site indicated the need of implementation the following aspects: Avoid civil and military aviation routes over the surroundings areas of the plant site, the consequences of external fire risk was minimized by increasing the surrounding area of high voltage power line, flammable substances transport risk were reduced by means of new route cambers and appropriate specific protections, intake air systems improvements to protect against smoke or heat from external fires.

The re-evaluation of man-induced events analysis for Atucha I NPP site is in progress. At present most of necessary date were compiled and the detailed evaluation is on-going also considering Atucha II according what is required in the mandatory documentation.

Requirement related to Man-induced Events for Atucha I and Embalse NPPs:

According to the man-induced events above mentioned and due to the plant is updating the information and the evaluations to met the mandatory documentation it was not necessary to issue specific requirements.

N° 69
CNS-Ref.-Art.: 17
Page of Report: 63
Chapter of Nat. Report: 17

The report states that "…the regulatory body (had) issued requirements to review the design parameters corresponding to earthquakes, extreme meteorological phenomena…" and that "The regulatory requirements were met."
What were the results from the review? Which specific requirements were met? Please elaborate on the criteria for evaluation. Please describe the licensing process relating to siting of nuclear installations.

Part a):

Earthquakes:
As was stated in the previous National Nuclear Safety Reports, after to meet the requirements about the seismic reevaluation of Embalse NPP carried out before the plant commissioning, the following additional requirements were met.

- Update and improve the program of plant response against seismic events: The plant prepared and implemented operating procedures (update and improvements) to assess actual plant physical damage and plant operational situation after the 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 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. The Operating Procedure was prepared, including the parameters for the Operating Basis Earthquake (i.e. SL-1 according to IAEA SG-50-S1 Rev.1).
- Implementation of a new digital seismic instrumentation: the plant implemented the definition, procurement, installation and commissioning of a new digital seismic instrumentation to provide data directly to plant operators for decision making and for immediate actions to check the DBE excedence. Besides, the parameters of the cumulative absolute velocity (CAV) were defined and the operating basis earthquake (OBE) given the corresponding values for Embalse Nuclear Plant. The instrumentation installed allows to record the seismic activity providing this information directly to the operator in the control room for decision making and immediate actions checking the operating basic earthquake (OBE).
- Need to perform a seismic Probabilistic Safety Assessment (in progress), the results of the seismic PSA required by the Regulatory Body could recommend some additional changes to the plant design.

Extreme meteorological phenomena:
The re-evaluation of tornadoes and severe storms hazard performed at Atucha 1 and Embalse NPP let to confirm that the tornadoes frequency distribution do not present differences within the uncertainty margin with the corresponding distribution as it were defined at the moment that the plant began to operate.

Besides, hydrological characteristics for the Atucha I site were re-evaluated considering the statistics from recent years in the Parana river and, as a consequence the extreme value of the flooding level was modified.

<u>Man-induced events:</u>
The results of man-induced events analysis carried out for Embalse NPP site indicated the need of implementation the following aspects: Avoid civil and military aviation routes over the surroundings areas of the plant site, the consequences of external fire risk was minimized by increasing the surrounding area of high voltage power line, flammable substances transport risk were reduced by means of new route cambers and appropriate specific protections, intake air systems improvements to protect against smoke or heat from external fires.

The re-evaluation of man-induced events analysis for Atucha I NPP site is in progress. At present most of necessary date were compiled and the detailed evaluation is on-going also considering Atucha II according what is required in the mandatory documentation.

Requirement related to Man-induced Events for Atucha I and Embalse NPPs:

According to the man-induced events above mentioned and due to the plant is updating the information and the evaluations to met the mandatory documentation it was not necessary to issue specific requirements.

<u>Part b)</u>: Requirements related to Protection against Earthquakes of Embalse NPP:

- Update and improve the program of plant response against seismic events.
- Implementation of a new digital seismic instrumentation to provide data directly to plant operators for decision making and immediate action to check the DBE exceedence
- Need to perform a seismic Probabilistic Safety Assessment

The criteria for evaluation applied was based in the National Regulatory Standards AR 3.10.1 “Protection against Earthquakes in NPPs”

Requirement related to Extreme meteorological phenomena Atucha I and Embalse NPPs:

- Reevaluation of Tornadoes and Severe Storms Hazards including transmission lines which are essential for the plant safety

The criteria for evaluation applied was based in the National Regulatory Standards AR 3.1.3 “Radiological Criteria Related to accidents in NPPs” that includes the corresponding probabilistic safety criteria

Requirement related to Man-induced Events for Atucha I and Embalse NPPs:

According to the man-induced events above mentioned and due to the plant is updating the information and the evaluations to met the mandatory documentation (Safety Report and PSR) and the External events IAEA Safety Standard NS-G-1.5, it was not necessary to issue specific requirements.

<u>Part c)</u>: The siting studies are part of the requirements the licensees shall comply with at the time they request a construction license. In the Argentine licensing process for a nuclear power plant, the licensing of a site is not required due to the site evaluation is included in the overall licensing process of NPPs (Standard AR 3.1.3). To grant the construction license must be demonstrated by the Responsible Organization that the plant design to be built complies with standards and other specific regulatory requirements for the selected site taking into account the NPP-site interaction.

The above mentioned studies includes the determination of the site characteristics that have influence on the effects of the plant operation on the environment, such as: meteorological and hydrological/ hydrogeological characteristics affecting the dispersion in the atmospheric and hydrological aquatic media, respectively, the population distribution and the regional uses of soil and water and man induced events including external missile impacts, aircraft crash and fire.

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Atucha II was designed approximately 30 years ago. Meanwhile the state of the art regarding safety requirements has changed, e.g. the scope of postulated accidents, consideration of beyond-design-basis accidents including severe accident management measures etc. How are these issues taken into account during the finalization of plant construction?

The main basis to grant the Atucha II Construction License were the considerations stated in the Preliminary Safety Analysis Report. Such report includes the verification of Standard AR 3.1.3 “Radiological Criteria Related to NPP Accidents”. The range of accidents covered for this standard are not classified discriminating design basis and beyond design basis accidents but considering that any accident can potentially occur and they has two parameters associated, its occurrence probability and its consequence. The consequences are measured in terms of individual doses.

The Regulatory Body requires that the Safety Report must be updated according to the state of the art at the licensing time, if applicable. However, in all cases, based on AR 3.1.3 the overall plant risk must be acceptable. In this sense, such update must include: the plant safety characteristics, the operating experience feedback from Atucha I, the operating experience applicable from other plants and updating coming from regulatory requirements.

It is important t mention that Atucha II has incorporated external experience and design improvements for other plants, mainly KWU reactors plants. It has also incorporated Atucha I experience in features which are particular to PHWR plants.

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CNS-Ref.-Art.: 18
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The second paragraph states that the regulatory body “…concluding that there is no objections to finalize the plant construction from the safety point of view”.
Considering that Atucha II was designed 30 years ago, please explain how Atucha II meets current regulatory and safety requirements.

The licensing process of NPPs considers 4 types of licenses: Construction, Commissioning, Operation and Decommissioning. The Atucha II NPP Construction License was granted by the Regulatory Body on July, 1981. At the beginning the construction activities were developed according the established schedule. However, since 1986 the construction was delayed and consequently the corresponding licensing process. Until now, the main regulatory activities have included:

- Preliminary Safety Analysis Report assessment.
- Preliminary Risk Analysis Report assessment.
- Responsible Organization, main suppliers, and main contractors audits.
- In-situ regulatory inspections during fabrication and installation of main components.
- Regulatory inspections to building site and components store.
- Mandatory documentation assessment.
- Brazilian/Argentinean regulatory bodies interaction to get experience from Angra II NPP licensing process, which had similar delay during the erection like the Atucha II NPP case.
- Safeguards design questionnaire report assessment.

It is important to mention that Atucha II has design safety features improved respect to Atucha I because it is similar to German 3rd and 4th generation plants and many aspects of Atucha I (unique German PHWR) operating experience were already considered in the Atucha II updated design such as:

- Significant operating events of Atucha I, in particular the most relevant event happened in 1988 “Fuel channel break”. Such event provoked reactor internal damages involving relevant corrective actions and design changes (see Annex 6, National Nuclear Safety Report 1998 “IAEA Safety Review Mission at Atucha I NPP”). This experience was considered to improve Atucha II original design.
- Reactor pressure vessel surveillance program.
- Reactor internal without “Stellite-6”.
- Emergency feedwater system (second heat sink for Atucha I) is already included in Atucha II original design.

However, there are additional features of Atucha II already considered coming from external experience mainly KWU reactors such as:

- TMI 2 designer implemented changes
- H2/D2 recombinators
- Improvements on computer operation support systems
- Updated safety analysis codes
- Environmental qualified of I&C, electrical and mechanical components and equipments

On the other hand, the Regulatory Body considers that it would be necessary to issue additional requirements due to last years improvements in the nuclear field associated with nuclear and radiological safety, safeguards and physical protection. The most relevant aspects to be considered are:

- Update of the Preliminary Safety Analysis Report in particular those issues regarding to plant risk assessment and compliance of the Standard AR 3.1.3 " Radiological Criteria Related to Accidents in NPPs".
- Update of external events analysis.
- Severe accidents assessment.
- Application of operating experience feedback from Atucha I.
- Detailed inspections on all components and equipments previously to their assembly.
- Update Safeguards documentation.
- Update Physical Protection Report.

There are some examples with delayed constructions of NPP around the world. In particular the Brazilian Angra II NPP, now in operation, can be mentioned due to its german technology like Atucha II, and delayed almost two decades according the original construction plan. The Argentinean Regulatory Body considered the Angra II licensing process to get the corresponding experience to be applied at Atucha II. There were no significant design changes in Angra II from the licensing process related to safety related components, equipments and systems. A detailed inspections on all components and equipments previously to their assembly was one of the main issues due to their long storage period exceeding the original schedule.

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Paragraph 1 states that "Maintenance activities for installed components and preserving … were carried out according with the manufacturers' specifications and guidelines."
Please explain how the licensee has taken advantage of experiences gained in preservation and long-term storage projects at other nuclear power plants that experienced similarly long construction/startup delays (e.g. Browns Ferry 1, Watts Bar).

The licensing process of NPPs considers 4 types of licenses: Construction, Commissioning, Operation and Decommissioning. The Atucha II NPP Construction License was granted by the Regulatory Body on July, 1981. At the beginning the construction activities were developed according the established schedule. However, since 1986 the construction was delayed and consequently the corresponding licensing process.
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Australia notes the decision that finalisation of construction of Atucha II NPP can proceed (18.3). Extensive information has been obtained from the operation of Atucha I and work to overcome design deficiencies (for example as reported in 6.1 of this report). To what practical extent is this improved understanding of the Atucha technology being incorporated into the 30 year old design of Atucha II?

The licensing process of NPPs considers 4 types of licenses: Construction, Commissioning, Operation and Decommissioning. The Atucha II NPP Construction License was granted by the Regulatory Body on July, 1981. At the beginning the construction activities were developed according the established schedule. However, since 1986 the construction was delayed and consequently the corresponding licensing process.

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What were the design changes considered and those that were actually installed for better safeguard controls of the dry-stored irradiated fuel elements.

The modifications were:

1- The re-verification tubes to install in a near future the remote monitoring system. This modification consisted of inter connecting these tubes of each silo in between and to a common collection system.

2- Sealing tubes were added (two) with an end to a "sealing box". These tubes allow the application of a dual containment system (metallic plus optical - cobra seals). This constitutes a modification to the silo because in the past, the verification tubes were also used to pass the cables or the optic fibers and in the upper level of the silo the seals were applied, also with a dual system.

All the tubes are made of metal.

From the safeguards point of view, these modifications are based on the need to adequate the design of the silos to the possible future transmission of data by remote monitoring and to the need to count on specific conduits only for sealing purposes (one tube for metallic seals and another one for optic seals). These modifications were introduced in the second project battery of silos built.

At present, preliminary tasks to increase the capacity of the dry storage of spent fuel elements have been performed. Currently, it has been proposed groups of 72 silos per stage (instead of forty silos per stage as it has been built up to now).

It is important to consider that all the modifications comply with the safety conditions with which this dry storage of spent fuel elements was licensed and foreseen in the original silos design.

N° 75
CNS-Ref.-Art.: 19
Page of Report: 67
Chapter of Nat. Report: 19

1) What is the extent of fill-up of liquid radwaste and solid radwaste storage capacities at the plants?
2) Is there a trend towards reduction in radwaste generation?
3) Do you undertake shipment of radwaste for final disposal/burial?

Radioactive waste management at NPP was described in the National Report to the Joint Convention on the Safety of spent fuel management and on the safety of radioactive waste management (November 2003).

Part 1): Low and very low level radioactive liquid waste are transitory stored at each NPP awaiting treatment on site. The storage period depends on the activity and treatment technology, and the filled-up portion is variable. Considering the different technologies used at each NPP for the treatment of low level radioactive liquid waste, herein below follows a description of such technologies :

Liquid radioactive wastes originated at ATUCHA I during operation and maintenance activities are collected in tanks and characterized to decide if they are dischargeable (according to pre-established procedures and within authorized constraints of discharge) or needs further concentration by evaporation. Concentrates and sludge from the cleanup of tanks are immobilized in cement matrixes for their conditioning in 200 liter drums.

Liquid radioactive waste originated at EMBALSE during operation and maintenance activities are retained by filters and ionic exchange resin beds, discharging into the environment the exhausted stream, based on pre-established procedures after characterization and within the authorized constrains of discharge.

Solid low level radioactive waste at both NPPs are classified as compactable, non-compactable and structural. Compactable solid waste are collected in plastic bags placed in predefined containers and further compacted in 200 L drums. Non-compactable and structural waste are disassembled and sectioned prior to the conditioning in 200 L drums. Such conditioned drums are stored temporarily at the facility.

Intermediate level radioactive solid wastes originated in the operation and maintenance activities of both NPPs, consist mainly of filters and spent ionic exchange resins. Such intermediate level radioactive solid wastes are stored at the facilities of each NPP.

In ATUCHA I, spent filters are stored at a temporary deposits of the facility. Such deposit consists of 8 boreholes of concrete walls, placed in a concrete platform limited by a fence. New boreholes are foreseen to be constructed as necessary. Spent resins are stored at two tanks of 15 m3 each, two tanks of 9 m3 each and in a concrete cubicle of 46 m3 . The remaining storage capacity is 25 m3.

At EMBALSE, filters from the ventilation and purification systems are stored in concrete underground cubicles and concrete boreholes with steel liner. In the underground cubicles, low level filters are stored, and higher level filters in boreholes. The total capacity is foreseen to be enough until the end of the facility life time. Spent resins are stored at two concrete tanks lined with epoxy resin, placed at the reactor building. The capacity of each tank is 260 m3, enough for the facility life time.

All storage facilities are placed at the NPP sites. Radioactive waste will be stored in such temporary facilities until new disposal facilities be constructed. Further details are presented at the National Report to the Joint Convention on the Safety of spent fuel management and on the safety of radioactive waste management (November 2003).

TABLE 1 – Storage capacity at ATUCHA I

WASTE TYPE	CNA I		
	Installed capacity (m^3)	Used (m^3) (at Dec. 2004)	Remaining storage capacity
Conditioned solids	As necessary	213 drums 42,6 m^3	
Spent-ion exchange resins	94	74,88	19
Filters	24	15,02 (3)	9
Structural	As necessary	354 ítem (*)	

(*) Fuel channel , control rod guide tubes, control rods and measurement devices

TABLE 2 – Storage capacity at EMBALSE

WASTE TYPE	EMBALSE		
	Installed capacity (m^3)	Used (m^3) (at Dec. 2004)	Remaining storage capacity
Conditioned solids	1344	253,2	81 %
Spent-ion exchange resins	520	181.53	61 %
Filters	135	23,475	61 %
Structural		29,15	

Part 2): The Responsible Organization policy establishes to minimize the impact on the environment as a result of its activity. Therefore, to keep to the minimum practicable the generation of radioactive waste is one of the main goals. To put in practice such policy an efficient and effective ALARA program has been implemented for both NPPs since starting operation.

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 in high radiation fields and waste generation.
- Compliance with segregation procedures of contaminated wastes.
- Damaged fuel elements are immediately withdrawn from the core.
- Operation power is usually maintained in a stable regime, with power ramps when needed, according to design.
- Personnel training in the application of radioactive waste management procedures.
- Measurement, classification, segregation and compactation of radioactive wastes.

In addition, the Responsible Organization has taken important actions to reduce the radioactive waste generated by the use of nuclear fuel. Presently in Atucha I low-enriched uranium fuel elements are being used and consequently, the generation of spent fuel elements has been reduced. Further, changes of the core channels bearing “stellite” reduced the amount of ^{60}Co activity in operational wastes. The generation of operational radioactive waste in Embalse was also reduced by changing the spent fuel storage way. Dry storage is air cooled different as required by wet storage (pools), that needs mechanical filters and ionic exchange resins to decontaminate the cooling water.

Part 3): Low level solid radioactive waste were previously stored at existing disposal facility located at Ezeiza Atomic Center (up to 1994 in Embalse and up to 1999 in Atucha I). Further details are presented in the National Report to the Joint Convention on the Safety of spent fuel management and on the safety of radioactive waste management (November 2003).

N° 76
CNS-Ref.-Art.: 19
Page of Report: 67-69
Chapter of Nat. Report: 19.8

The report includes no information on article 19, clause viii. Please, present the fulfilment of clause VIII.

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N° 77
CNS-Ref.-Art.: 19
Page of Report: 68
Chapter of Nat. Report: 19.2

Have strategic plans and specific actions been implemented and staff trained to cope with the "six scenarios" for plant damage states?

The first stage of the Severe Accident Management Program, for Atucha-I Nuclear Power Plant (CNA-I) has just finished. The times and main physical conditions corresponding to each of the six scenarios considered at the onset of the core damage has been identified. In spite of some potential strategies have been identified, the Responsible Organization is assessing the effectiveness of each one of such strategies, equipment needs and human performances to elaborate the corresponding procedures and the staff training program.

Examples of accident management strategies considered are the following:

- Specific small LOCAs with ECCS unavailable: strategy to restore light water to the reactor coolant system through the inventory control system. This strategy is accomplished by water supply to the inventory control system storage tank.
- Station blackout: strategy to maintain a long-term reactor heat removal using the second heat sink system (SHS) avoiding air entrance from boron injection system, which is operated by exhaustion of batteries. This strategy is accomplished by manual disconnection of the boron injection system before the air entrance.
- Station blackout / feedwater system failure : strategy to ensure adequate long-term reactor heat removal capability through SHS available. This strategy is accomplished by using the SHS injection lines; and providing backup for water supply to SHS from others reservoirs.

N° 78
CNS-Ref.-Art.: 19
Page of Report: 67
Chapter of Nat. Report: 19.1

Do operator reports on the evaluation of international operating experience contain information on which events in foreign plants have been screened?.
How are “low level events” and “near misses” reported by operators?

Part a): The Operator reports on the evaluation of international operating experience includes information related with the international events selected after an adequate screening. These reports also include root and direct causes and corrective actions taken.

According to the requirement issued related to Operating Experience Feedback Program, the Operator send to the Regulatory Body a complete list of analyzed and screened events.

Part b): A quarterly report is sent to the Regulatory Body by the Responsible Organization containing a list of “low level events” and “near misses”. In addition, the Regulatory Body, could require, if necessary, more information about specific events to verify precursors, event recurrence and lesson learnt.

N° 79
CNS-Ref.-Art.: 19
Page of Report: 69
Chapter of Nat. Report: 19.3

Please provide information as to how the findings of the WANO peer reviews of the responsible organization do compare with the reviews of the regulatory body. Also, please indicate the significant discrepancies between the reviews and provide specific examples, if any.

No significant discrepancies between WANO and regulatory body reviews were found. Findings coming from WANO peer reviews carried out in both NPPs such as improvements on: ALARA strategies, operational feedback experiences applications and safety culture reinforcement are in agreement with regulatory requirements.

N° 80
CNS-Ref.-Art.: 19
Page of Report: 69
Chapter of Nat. Report: 19.3

The report only mentions progress made in the field of operational experience feedback and accident management since the previous report. Doest this mean that since 2002 no progress was observe in the other fields of Article 19 of the Convention (i.e.: revision of operational limits and conditions, maintenance, technical support, incident reporting, generation of radioactive waste)?

Additional progress in different areas are not mentioned in the report by consider them as very specific and minor changes.

Considerable progress has been made in Atucha I regarding the review of operational limits and conditions, technical support and improved assessments during the backfitting identifying safer operation boundaries as it indicated in Article 6.

In the case of Embalse NPP several changes in the maintenance area were carried out, in particular all safety related components surveillance test were reviewed with the purpose of improving the acceptance criteria tests in the surveillance program. Additionally changes in the mandatory documentation regarding components test and preventive maintenance frequencies were carried out in order to improve practices in terms of operator doses. These changes are based on the assessment of maintenance history records, safety system reliability reports, PSA reports and repetitive test records.

In the area of generation of radioactive waste additional efforts were performed in both plants to improve the planning of the activities that involve significant waste generation and/or individual doses. Better segregation procedures has been put in practice for low and intermediate solid waste. Additional personnel training in the procedures of radioactive waste management were implemented.

As mentioned within the Quality Management System implemented in the Regulatory Body several procedures have been implemented covering areas such as incident reporting system, operating experience feedback, regulatory inspections and evaluations that let to improve the effectiveness and efficacy of the control activities.

N° 81
CNS-Ref.-Art.: 19
Page of Report: 67
Chapter of Nat. Report: 19

What are Argentina's plans on decommissioning and the treatment of nuclear waste? Are there any storage facilities in use or under construction?

Decommissioning:

In accordance with Nuclear Law No. 24804 Argentina's responsibility for NPPs decommissioning activities rest on the National Atomic Energy Commission (CNEA). Decree 1390/98, related to Nuclear Law, establishes that the Responsible Organization of NPPs operation (NASA), shall notify to both CNEA and the Regulatory Body the end of commercial operation at least one year in advance.

The Nuclear Installations Dismantling Program established by CNEA defines the organization and the following activities that shall be performed on this matter:

Planning and control of dismantling and decommissioning management activities of Argentinean major nuclear installations.

Coordination of the development of specific technologies.

Training of personnel.

Promotion of international agreements.

Current tasks are the following:

Development of a Quality Management System.

Definition of dismantling and decommissioning alternatives for Argentinean research reactors.

Planning and dismantling cost of the two nuclear power plants in operation.

In December 2001 CNEA and NASA agreed to make a joint analysis of the transition stages after the decommissioning of Atucha I. Later this analysis will be carried out for Embalse.

The scope of the current study is to determine the technical actions that shall be taken for decommissioning, considering different dismantling alternatives, as well as resources (personnel, energy, maintenance, spares, works) and the expected radioactive wastes. The study shall continue with the definition of the dismantling and decommissioning strategy in accordance with the Radioactive Waste Management Strategic Plan.

Radioactive Waste Management

The legal framework applicable to radioactive waste is commensurate with the provisions of the National Constitution and with the legislation adopted by the National Congress by Law Nº 24804 which regulates the nuclear activity and other activities, and Law Nº 25018 which lays down the Radioactive Waste Management Regime.

Argentina has developed a legal and regulatory structure which complies with the safety provisions established in the Joint Convention. CNEA is the Responsible Organization for spent fuel management and radioactive waste management. Provisions have been adopted for NPPs spent fuel management, their interim storage facilities up to a decision of their final disposal as radioactive waste is taken.

Radioactive waste management policy

Radioactive waste management policy specified in the National Report to the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management (November, 2003) is contained in the following statements:

- To manage radioactive waste originating from domestic nuclear energy applications, including wastes from the decommissioning of related facilities.
- The definition of responsibilities to carry out waste management activities, and essentially the long term surveillance and institutional control required by different final disposal systems used.

- The objective of such management activities be performed safely, ensuring the protection and the rights of present and future generations and the environment.
- The development of an Strategic Plan which is periodically reviewed, authorized and audited by the National Congress.
- To establish a proper mechanism to obtain and manage the necessary financial resources to comply with the obligations arising from the performance of the assigned responsibilities, considering that many of them imply deferred costs.
- Maintenance of a recording and information system which provides a total knowledge and control of inventories of radioactive waste generated and to be generated from all nuclear activities.
- Development of a public communication and information program.

Spent fuel management policy

Spent fuel management policy specified in the National Report to the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management (November,2003) is contained in the following paragraphs:

Under the terms of Article 2, paragraph g, of the National Law of Nuclear Activity Nº 24804, the Argentine Government, through CNEA, exercises the state ownership of the special radioactive fission material contained in spent fuels originating from the operation of Nuclear Power Plants and from experimental, research and/or production reactors.

The decision on the strategy to be adopted, either for their reprocessing in order to separate and recover the fissile material or its final direct disposal as radioactive waste, is related to political, financial, energy, environmental and safeguard factors because spent fuels are considered a potential energy resource.

In line with the Radioactive Waste Management Strategic Plan, the decision on the use or not of the fissile material contained in spent fuels has been postponed. In the meantime, research and development programs are being organized in order to have the necessary technical and human resources to implement the decision adopted.

N° 82
CNS-Ref.-Art.: 19
Page of Report: 70
Chapter of Nat. Report: Planned Activities

The report does not contain any indication about planned activities to improve safety: would it be possible to present the main objectives of the regulator and the operators in that field?

- Regarding Atucha I the recent backfitting program has provided significant improvements to the design and operational plant safety (see Article 6).
- Measures to improve the safety at Embalse NPP are being planned by using current CANDU community safety improvements, specific plant PSA results and specific safety assessments as a documents basis.
- Application of PSAs results coming from “Sources different than reactor core”, “Shutdown and Low Power” , “Fire Risk” and “Seismic” would produce safety improvement in both NPPs.
- Operating Experience Feedback applied at Atucha I must be improved. A dedicated regulatory audit is being planned for 2005.Embalse NPP refurbishment (PLIM + PLEX) is being analyzed by the Responsible Organization.
- An IAEA Design Review Mission is being considered to review safety aspects of Atucha II plant design.
- Peer Review WANO missions are planned for both NPPs by 2006/7.
- To reinforce the Ageing Management Program application
- A backfitting program has not been implemented for Embalse NPP yet. Such activity is being planned by using current CANDU safety improvements, PSA results and specific safety assessments as a documents basis.
- Both the Responsible Organization and the Regulatory Body agreed that operational safety issues are part of the improving activities for both NPPs, in particular in the following areas: operating experience feedback, maintenance, ageing, safety culture and severe accidents.

N° 84
CNS-Ref.-Art.: General
Page of Report:
Chapter of Nat. Report:

The report is informative, precise and presents all relevant data. Although all aspects are thoroughly discussed there are discrepancies between the list of contents and the individual chapters in each article (compare index for Art. 8).

N° 85
CNS-Ref.-Art.: General
Page of Report: 13
Chapter of Nat. Report: 6.1.4

Which improvement in reduction of total CDF was achieved for Atucha I by the backfitting program?

After implementation of the backfitting program the core damage frequency of Atucha I was reduced one order of magnitude reaching a value below 10^{-04} / year without recovery actions considerations.

The design changes were focused on the sequences that implied deficiencies and failures in the residual heat removal system. Previously to the backfitting implementation, the CDF contribution of sequences like “Loss of feedwater system” and “Loss of circulating water condenser” were highly reduced due to the installation of the second heat sink system. Additionally, design changes in the main steam system improved significantly the reliability of the relief valves. The contribution of blackout sequences were also improved due to the connection of additional emergency diesel generators as a back-up emergency power supply.

In addition, following with the continuous improvement concept, a more detailed modeling of sequences using both better codes and an improved nodalization recently performed shown that some plant damage states initially considered in the Atucha I PSA were conservative, in particular those related to the operator time-windows to take corrective actions. This new development will allow to update the Atucha I PSA and obtain new results that will impact in the core damage frequency.

N° 86
CNS-Ref.-Art.: General
Page of Report: 7
Chapter of Nat. Report:

Argentina has chosen not to address it its third report the aspects that remain unchanged from the previous report. However the fact that the regulation has not changed does not necessarily mean that there has been no progress in its implementation. In that sense a more self-standing report would have been appreciated which would have allowed to clearly highlight the trends, progress and difficulties in complying with the regulation.

This National Report was elaborated according to “Guidelines Regarding National Reports under the Convention on Nuclear Safety lineaments” in order to avoid repetitions. Therefore, the third Nuclear Safety Report describes and details all the progress and actions related with the Nuclear Safety Convention obligations carried out in our NPPs since the second Nuclear Safety Report. Additionally, to facilitate the understanding and follow up process, this report includes the conclusions to the two earlier reports and the full list of questions and answers to the second. However, Argentina will consider to include in the next reports the essential information of the previous report to provide a more self-standing report.

N° 87
CNS-Ref.-Art.: General
Page of Report: 1
Chapter of Nat. Report:

The reports reviewed by France in view of the third peer-review meeting were all examined according to a standard list of issues derived from the obligations of the Convention. If an

issue appeared to be covered in an incomplete way by the report of a Contracting Party, this led to a question or comment. However France recognizes that the corresponding information may be available in other existing documents.

N° 88
CNS-REF.-ART.: GENERAL
PAGE OF REPORT: 7
CHAPTER OF NAT. REPORT: INTRODUCTION

What do you see as major challenges for the future and how are you addressing them?

The main challenges considered in the nuclear sector are the following:

- Atucha II construction, full restarting
- Plants life extension program
- Ageing aspects
- Maintaining nuclear competence in both the Responsible Organization and the Regulatory Body the corresponding criteria to face with the future challenges.

N° 89
CNS-REF.-ART.: GENERAL
PAGE OF REPORT: 67
CHAPTER OF NAT. REPORT: 19.1

The intention of the convention of Nuclear safety is in some sense understood to improve safety of NPPs by backfitting event/failure experience of other countries, could you explain the policy and present/future perspective of information disclosure to public?

The Regulatory Body and the Responsible Organization are aware of the relevance of the nuclear activities information to public. The policy related to nuclear activities information to the public includes as a main objective to inform about the different activities that are performed. To met such objective the following actions are carried out:

- Backfitting event/failures experience: The Operating Experience Feedback program applied by the Responsible Organization includes to share the information and lesson learnt from CANDU Owners Group (COG), WANO, IAEA-IRS, IAEA-INES, specific meetings and workshops. The Regulatory Body share the information with IAEA-IRS, CANDU Senior Regulators Group, Bilateral Regulatory agreements, Ibero-American Forum of Regulators, IAEA-INES, specific meetings and workshops.
- To organize specific meetings in all the country with the community participation.
- Participation of the community in NPP emergency drills. In particular, neighbours and schools personnel participates in the drills.
- To inform through dedicated web sites:
  - Regulatory Body: http://www.arn.gov.ar
  - Responsible Organization: http://www.na-sa.com.ar
- Press releases: about concerns related to nuclear activities, in particular in case of significant events.

<u>On a regular basis the following public institutional issues are released:</u>

- *Regulatory Standards*.
- *Regulatory Annual Report*: It is sent to National Congress and includes the regulatory activities and facts performed in that period. Annual reports summarize the main regulation and control activities regarding radiological and nuclear safety, safeguards and physical protection which are carried out each year at all facilities and for all practices involving ionizing radiation throughout the country. This report, regularly sent to Parliament since 1997, describes the Argentine regulatory system, the facilities under control and the main regulatory activities performed by the ARN during the period from January 1 to December 31 each year. Annexes containing all licences, operating authorizations and transport certificates issued and

inspections performed during the year at medical, industrial and research and teaching facilities are included.

- *Technical Reports*: contain all works published and/or submitted to congresses by the ARN's different working groups in the field of radiological and nuclear safety, safeguards and physical protection. They also include works performed under agreements between the ARN and universities or other local and foreign organizations. The Technical Reports are edited on a yearly basis since the creation of the Regulatory Authority.
- *National Nuclear Safety Reports*.

N° 90
CNS-Ref.-Art.: General
Page of Report:
Chapter of Nat. Report:

Australia notes that Argentina has again presented an updated report which, for full appreciation, requires access to the two previous reports. Argentina defended this approach in response to a comment on its second report (Question No. 79) as being in accord with the Guidelines Regarding National Reports. The Argentine report also, very usefully, includes the conclusions to the two earlier reports and the full list of questions and answers to the second. It would, however, be helpful to have included in the current report some brief summary of the essential information from earlier reports.

Argentina will take into account this comment for the next National Nuclear Safety Reports.

N° 91
CNS-Ref.-Art.: 8.2
Page of Report: 25
Chapter of Nat. Report: 8

It has continued to participate in the ..... And the Network of Regulators of Countries with Small Nuclear Programs. It should be considered as a good practice to ensure maintenance of the competence. Such networks should be established also for other regions.

Argentina agrees with the comment.

ANNEX V

MAIN TECHNICAL FEATURES OF THE ARGENTINE NUCLEAR POWER PLANTS IN OPERATION

V.1. ATUCHA I NUCLEAR POWER PLANT

V.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 l is located by the right side of Paraná River, same 7 km from Lima, Province of Buenos Aires, and near 100 km to the north-west of Buenos Aires city. ***Figure V.1-1*** shows its geographic location.

The owner of CNA I is Nucleoeléctrica Argentina SA, 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 V.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 M Wt at that time.

Nevertheless, afterwards, the station suffered two essential modifications that improved its performance:

- In 1977 bits electric power was increased to 357 M We (335 M We net) and, correspondingly, its thermal power to 1179 MM.
- 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 1, 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.

V.1.2. OVERALL PLANT LAYOUT

The overall layout of Atucha 1 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 V.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.

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 he nuclear power plant.

The reactor building ***(Figure V.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: 50m) is constructed as a pressure vessel and designed for the maximum pressure associated with the worst event which has to be taken into account.

The ancillary and low-pressure- leading auxiliary systems and components (e.g. residual heat removal system, safety injection system or heavy water storage system) are accommodated in the reactor building annulus, the annular space between the cylindrical part of the concrete containment and the spherical steel containment.

A special ventilation system for the annulus ensures that even under accident conditions small radioactive leakages from the containment are retained by charcoal filters, thus preventing any radiation hazards to the environment.

The low-level arrangement of the reactor building and the heavy internal concrete structures, as well as the massive outer concrete shield provide good protection against seismic and other external loads. At the same time, they subdivide the interior of the reactor building into operating and plant compartments. Due to special ventilation systems the former is accessible for Inspection and maintenance work during reactor operation without restriction and without any special protective measures.

The plant compartments for reactor, steam generators and pumps are provided with removable covers, so that all heavy components can be serviced by the polar crane.

The systems necessary for on-load refuelling are also housed in the containment structure.

The reactor auxiliary building adjoins the reactor building, and surrounds a part of it, thus allowing short connections to the equipment located in the reactor building annulus.

On top of the building the vent stack is situated.

The fuel storage building is linked with the reactor building by the fuel transfer system. Personnel access is possible from the reactor auxiliary building. The spent fuel assemblies are transferred from the reactor to the fuel storage pools with the aid of the fuel transport system, consisting of. refuelling machine, tilter, transfer tube, tilting device and manipulating bridges. The new fuel assemblies are supplied to the reactor in the reverse way.

The turbine building is of a two-bay design. Its dimensions are governed to a large extent by the dimensions of the turbine generator and its auxiliaries. The main bay houses the turbine generator set and the feedwater tank.

The lower bay houses the condenser, the feedwater pumps and other equipment associated with the steam/feedwater cycle. All these compartments are free of radioactive media. Besides the turbine building contains the emergency diesel generator and the high-voltage station service transformer.

The main steam lines coming from the reactor building enter the turbine building along the shortest route leading to the area of the high pressure casing of the turbine, where the main steam flows through the steam strainers into the high pressure turbine. Vertical moisture separators are installed on both sides of the high-pressure casing.

The turbine operating floor is clear of off pumps and piping so that floor' space is available to set down components removed when the turbine generator unit is opened yap for major overhauls.

Floor space has been left clear for a clamping fixture. The feedwater tank with the deaerator is installed on the level of the turbine-operating floor. The feedwater pump units and the start-up and shutdown pumps are installed underneath.

An overhead travelling crane for transporting and erecting plant components is installed in the turbine building.

The controlled access to the "controlled area" is located fn the same floor of the switchgear building as the control room.

Ventilation of individual floors is provided by the ventilation systems through redundant intake and exhaust air shafts and smokes vents.

This is accomplished by the extreme leak-tightness of the primary systems, adequate shielding and physical separation of equipment, and by the ventilation which enforces a definite continuous airflow from rooms with lower activity to rooms with higher activity.

V.1.3. CNA-I MAIN SYSTEMS

In what follows the main safety and process systems that are part of the station, are briefly summarised.

V.1.3.1 Reactor

The reactor ***(Figure V.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 M Wt.

The reactor core is approximately cylindrical in shape and consists of 252 natural and 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 bum-up to be attained. Furthermore, the connection between the reactor coolant system and the moderator system permits the use of common auxiliary systems to maintain the necessary water quality. The number of auxiliary systems can therefore be reduced to a minimum.

In order to control reactivity, and thus the power output of the reactor, various methods are applied. The reactor contains 24 "black" (absorbers made of hafnium) and 5 "grey" (steel) control rods. The control rods are used to control reactivity and power distribution, to compensate the build-up of xenon poisoning after a reactor power reduction, to provide damping of azimuthal xenon oscillations, and to shut down the reactor.

In addition to the control rods, reactivity control is provided by the boric acid dosing system. The injection or extraction of boric acid serves to compensate slow reactivity changes due to bum-up during the first period of operation. Extraction of the boric acid is performed by anion exchangers.

Additionally, a boron injection system, as a second independent shutdown system is provided, which injects boric acid into the moderator.

In addition to these reactivity control systems, reactivity can also be controlled by varying the moderator temperature within a certain range, which is advantageous for some operating modes.

The reactor pressure vessel constitutes the pressure boundary of the reactor core and encloses core components and reactor pressure vessel internals. The reactor pressure vessel consists of a lower part, the closure head and the studs and nuts which connect both sections. The connection is made leak-tight by means of a welded lip seal.

Most of the reactor pressure vessel internals form the structure of the reactor core. The moderator tank accommodates all core components, separates the moderator from the coolant, and, in conjunction with the reactor pressure vessel, forms the annulus for the in-flowing coolant. The moderator tank bottom serves as the lower fixing level for the coolant channels and the control rod guide tubes. The moderator tank shell serves as thermal shielding.

The moderator tank closure head forms the upper plenum for the reactor coolant leaving the coolant channel slots. The closure head and the moderator tank jointly form a unit which keeps the coolant channels and the control rod guide tubes in position firmly and without displacement during all operating modes, as well as during postulated accidents. The moderator tank and its closure head are suspended from the flange of the reactor pressure vessel and are field tightly in position by the pressure vessel closure head.

The coolant channels consist of vertically-arranged tubes which contain the fuel bundle columns, direct the reactor coolant flow and separate the reactor coolant from the surrounding moderator.

The reactor coolant flows inside the coolant channels in an upward direction. After passing through the fuel assembly, it leaves the coolant channel through slots and enters the upper plenum formed by the moderator tank closure head.

The coolant channel closure head, together with the coolant closure plug, forms the pressure-tight cap of the coolant channel. It can be opened by the refuelling machine during reactor operation in order to exchange the fuel bundle column located inside the coolant channel.

The moderator piping serves for supply, distribution and extraction of the moderator inside the moderator tank. The moderator piping essentially encompasses down-comers, the sparger ring on the moderator tank bottom, and the suction boxes with nozzles in the moderator tank closure head.

The moderator flows downwards through the down-comers to the sparger ring, where it is distributed at the moderator tank bottom. After rising and heat-up in the moderator tank, the moderator flows to the suction boxes and leaves the moderator tank through two nozzles.

V.1.3.2 Reactor coolant system and moderator system

The reactor coolant system ***(Figure V.1-6)*** removes the heat generated in the reactor core and transfers it via the steam generators to the turbine generator plant.

The system is structured similar to that of a pressurised light water reactor and consists of two identical loops, each comprising a steam generator, a reactor coolant pump and the interconnecting piping, as well as one common pressurizer.

The heat is carried by the reactor coolant, which flows from the reactor pressure vessel to the steam generators, where it is cooled down and then pumped back to the reactor pressure vessel by the reactor coolant pumps.

The pressurizer system is connected to one reactor coolant loop and basically comprises the pressurizer with the electric heaters, the surge line, the spray lines with valves, and the safety valves.

Besides pressure control by sprays in the pressurizer, protection against over-pressure in the reactor coolant system is provided in accordance with international codes for pressure vessels and systems. Protection is afforded by independent, self-actuating safety valves.

When the safety valves open, the steam discharged from the pressurizer is directed into the pressurizer relief system, where it is condensed to water.

The moderator system consists of two identical loops operating in parallel. Each loop comprises a moderator cooler, a moderator pump, and the interconnecting piping with valves.

The moderator system performs various functions depending on the operating mode of the reactor.

During normal operation the moderator system maintains the moderator at a lower temperature than that of the reactor coolant. The moderator leaves the top of the moderator tank flows to the moderator pumps, is pumped to the moderator coolers and flows back to the bottom of the moderator tank. The heat transferred in the moderator coolers is used for pre-heating the feedwater.

For residual heat removal the moderator system is switched over to the residual heat removal position by means of the moderator valves. Under this mode of operation, the moderator is extracted from bottom of the moderator tank by the moderator pumps and fed into the cold legs of the reactor coolant loops, and also directly into the reactor coolant inlet annulus of the reactor pressure vessel via the moderator coolers. The moderator system forms the first link of the residual heat removal chain. The residual heat is transferred from the moderator system to the residual heat removal system and thin to the service cooling water system.

During emergency core cooling the moderator serves as a high-pressure core re-flooding and cooling system. The emergency core cooling position is similar to that of the residual heat removal, but additionally, water is injected into the hot legs of the reactor coolant loops and into the upper plenum of the reactor pressure vessel. The residual heat removal chain connected to the moderator coolers during emergency core cooling is the same as during residual heat removal.

An essential feature of the moderator system together with the residual heat removal system is the allowing of the hot shutdown condition of the reactor to be maintained for as long as required, or the cool-down at a pre-set gradient, as well as emergency core cooling without main steam blow-off and thus without an additional heat sink.

All systems of the residual heat removal chain are of a consistent two-loop design. The residual heat removal system acts as a barrier between the active moderator and the service cooling water and prevents the escape of radioactivity into the service cooling water in the event of leakages in the moderator coolers.

V.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 p[ant operation. The refuelling procedure is carried out by a single refuelling machine. The fuel assembly transport system is located in the reactor building and in the fuel pool building. The main items of the fuel transport system are: refuelling machine, tilter with supporting structure, fuel transfer tube, fuel pools, and the corresponding auxiliary systems and maintenance installations.

The refuelling procedure is fully automated and monitored from the control room.

The refuelling machine is moved from a maintenance position in the refuelling machine maintenance room, by remote control, to a previously selected coolant channel position in the reactor well in which the machine is centered. The seat-on seal is pressed on to the coolant channel closure head by the dead weight of the refuelling machine to form a watertight seal between the machine and the coolant channel. Pressure equalisation takes place between the refuelling machine and the reactor before opening the isolation valve of the refuelling machine and opening the coolant channel closure. Following this, the fuel bundle column is withdrawn into an empty position in the refuelling machine magazine. The magazine is then rotated in such a way that a fuel bundle column with a partially burnt-up fuel assembly or with a new fuel assembly is positioned above the open coolant channel. This fuel bundle column is lowered into the coolant channel position and the coolant channel closure is locked again. After closing the isolation valve of the refuelling machine a check for leak-tight closure is performed. Then the refuelling machine is removed from the reactor pressure vessel and positioned above the vertically arranged tilter. The tiIter has the following functions in the indicated sequence:

- Take-over of the fuel bundle column with the spent fuel assembly
- Removal of the decay heat by cooling with heavy water
- Drying and cooling the spent fuel assembly with gas
- Flooding and cooling of the tiIter with heavy water
- Tilting to the horizontal position and connecting with the fuel transfer tube
- Transfer of the fuel assembly into the fuel transfer tube.

When a new fuel bundle column is transported from the fuel pool building into the filter 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.

V.1.3.4 Reactor auxiliary and ancillary systems

The auxiliary systems are basically organised 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 minimised. 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.

V.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.

V.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.

V.1.3.7 Electric power system

The Atucha I nuclear power plant has two physically independent grid connections ***(Figure V.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 6.6kV) 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 220kV and 132kV 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

{6.6kV 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 220kV 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.

V.1.3.8 Safety systems

The safety philosophy, on which the design is based, fulfils, In all conceivable plant conditions, the following basic requirements:

- The reactor can be safely shut down and kept shut down over prolonged periods (the decay heat can be reliably removed)
- Any release of radioactivity is within the limits established by the radiation protection regulations.

In order to meet these requirements, safety measures against damage to the systems or components are provided. Safety measures can be classified in three safety levels according to the possible plant conditions:

Components and systems necessary for normal operation (including startup, partial load and full load operation, load changes, shutdown) are of such design as to preclude failure. The safety measures provided are:

- Conservative and careful design
- Stringent quality assurance and control
- Regular examinations and inspections.

According to general engineering experience, it must be considered that systems and components can fail during their service life despite adequately high quality. It is assumed that operational disturbances (e.g. reactor coolant pump failure, load rejection) can occur. In order to prevent faults and operational disturbances and to mitigate their consequences the following safety measures are provided:

- Inherently safe Operational characteristics
- Alarm annunciation
- Reactor protection limitation.

Despite the safety measures of the first and second safety levels, theoretically assumed accidents are postulated. In order to prevent these accidents and to mitigate their consequences, safety systems are provided. The design of the safety systems is based on the assumption that parts of the safety systems (sub-systems) can fail simultaneously with the accident. As a consequence, safety systems are of redundant design.

The basic safety systems provided are:

- Fast Reactor Shutdown System

- Emergency Core Cooling System
- Containment System
- Emergency Electric Power System (safety related system)

In order to protect the environment against the release of radioactivity, the following radioactivity barriers are provided as passive safety measures:

- The fuel matrix of the uranium dioxide pellets
- The seal welded cladding tubes enclosing the fuel
- The closed and seal-welded reactor coolant system and moderator system
- The full-pressure gas-tight steel containment structure
- The concrete secondary shield.

The components of the radioactivity barriers act according to their mechanical properties, without auxiliary energy. In case of damage to one of these barriers, the next one will act and thus retain the radioactivity.

The accidents considered in the plant design are the plant internal and external accidents. The internal accidents are, above all, loss of coolant accidents (LOCA), with the whole spectrum of pipe ruptures including the break of the largest connection pipe to the reactor coolant loops or to the moderator system. The external accidents considered are aeroplane crash, explosion pressure wave, floods, tornadoes, etc.

In order to meet the safety requirements even during the considered internal and external accidents, the following design principles were established:

- Multiplicity of safety features
- Redundancy of safety systems and of their auxiliary systems
- Diversity of important parts of the reactor protection system
- Physical separation and/or protection by concrete walls of the redundant sub-systems
- Protection of safety systems against external accidents
- Periodic testing of safety systems.

The task of the safety systems is to prevent any damage to the radioactivity barriers during operational malfunctions and during accidents in order to fulfil the safety philosophy requirements.

The fast reactor shutdown safety system consists of two separate sub-systems: the shutdown control rod system (first independent shutdown system) and the boron injection system (second independent shutdown system). The emergency core cooling safety system consists of the following basic sub-systems: the moderator system, the residual heat removal system, the service cooling water system for the secured plant, the nuclear component cooling system and the safety injection system.

The containment safety system consists of several basic sub-systems: the concrete containment, the steel containment, the containment isolation system and the reactor building annulus air extraction system.

The safety systems are supported by a high degree of quality assurance and quality control measures, regular inspections during operation of the plant and in-service inspection programs. Through these measures, a high safety standard can be ensured.

V.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 M We
Authorised thermal power	1179 MWt

Reactor Core Data	
Type of fuel	Slightly enriched uranium (0.850 weight)
Shape of fuel assembly	37 - rod cluster
Number of coolant channels or fuel assemblies	252
Cladding material	Zircaloy 4
Fuel assemblies length	6180 mm
Refuelling	On load
Coolant and moderator	Heavy water

Thermal and Hydraulic Data	
Pressure at reactor vessel inlet	12.2 MPa
Pressure at reactor vessel outlet	11.6 MPa
Coolant channel inlet temperature	264°C
Coolant channel outlet temperature	303.3°C
Maximum temperature on the fuel assembly cladding surface	325°C
Coolant flow in coolant channels	20210 t/h
Average coolant speed in central channel	9 m/s
Mean heat-flux density	67.7 W/cm^2
Average specific thermal power of fuel roofs	232 W/cm
Heavy water concentration	99.75 (weight)

Steam and Power Conversion System Data	
Live steam pressure at steam generator outlet	4.46 MPa
Live steam temperature at steam generator outlet	254.9°C
Live steam flow	1856 t/h
Live steam moisture	0.3 %
Turbine rated speed	3000 rpm
Condenser pressure	4.56 kPa
Cooling water inlet temperature of condenser	22°C
Cooling water flow of condenser	62500 m^3/h
Generator apparent power	425 MVA
Generator power factor	0.8
Generator voltage	21 kV
Generator transformer rated power	400 MVA
Generator transformer transformation ratio	21 kV / 245 kV
High-voltage station service transformer rated power	35 / 20 / 20 MVA
High-voltage station service transformer transformation ratio	21 kV / 6.95 kV
Generator off-site system transformer rated power	35 / 20 / 20 MVA
Oft-site system transformer transformation ratio.	132 kV / 6.95 kV

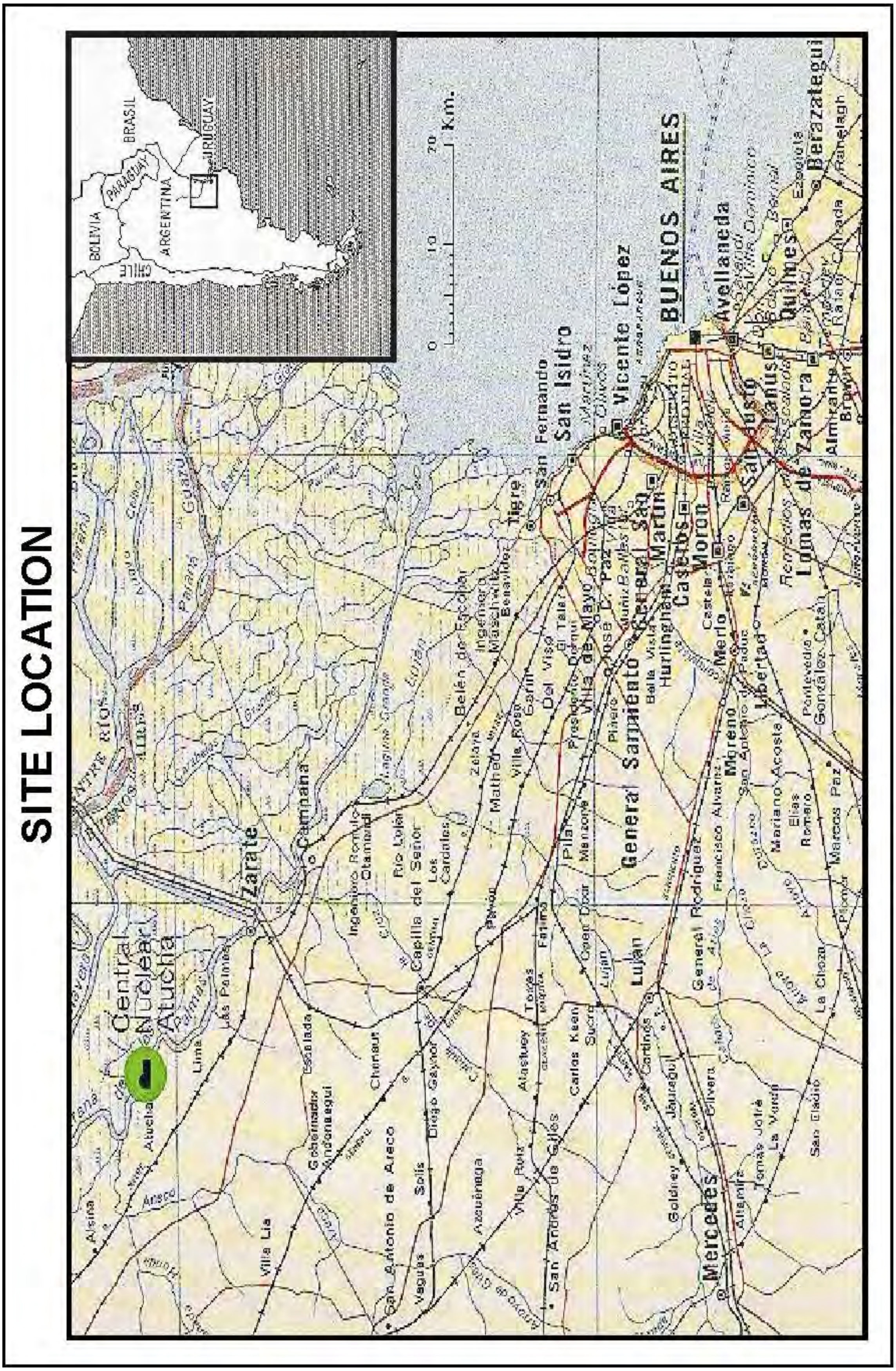


Figure V.1-1 - Atucha I Nuclear Power Plant - Geographic Location

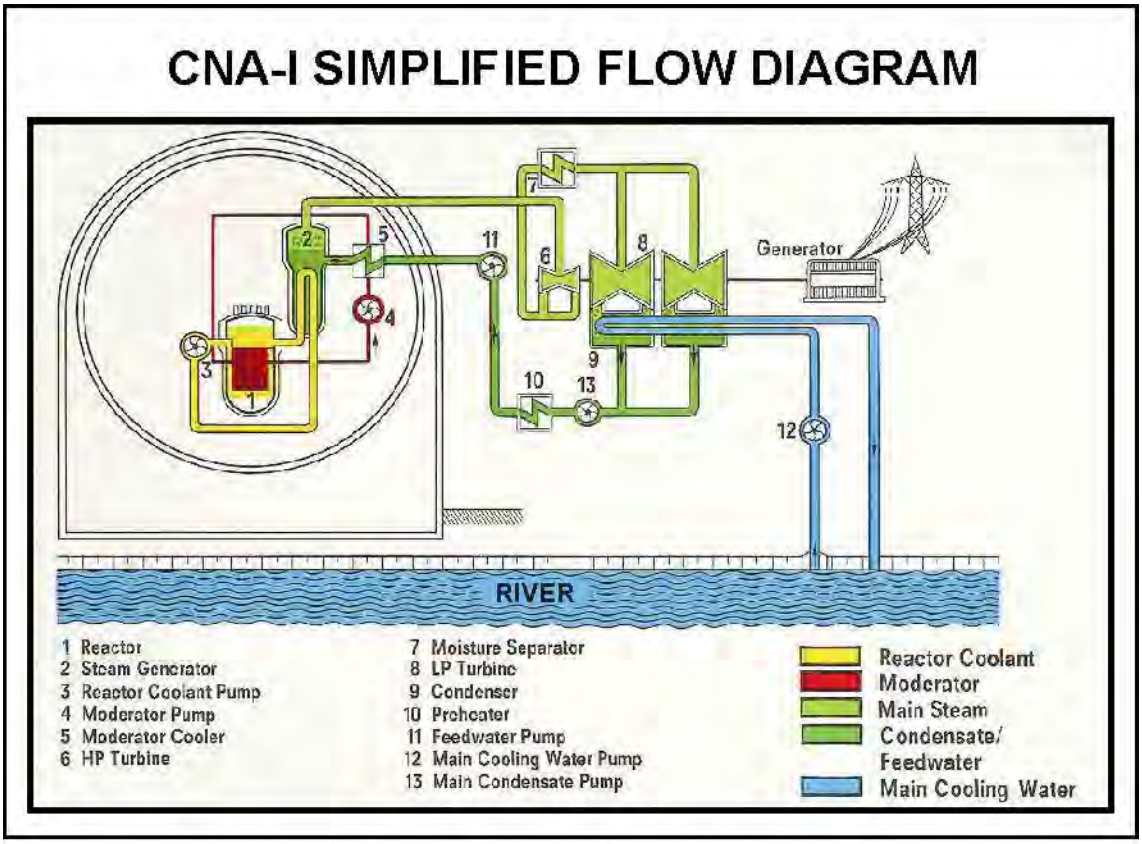


Figure V.1-2 - Atucha I Nuclear Power Plant - Main Systems

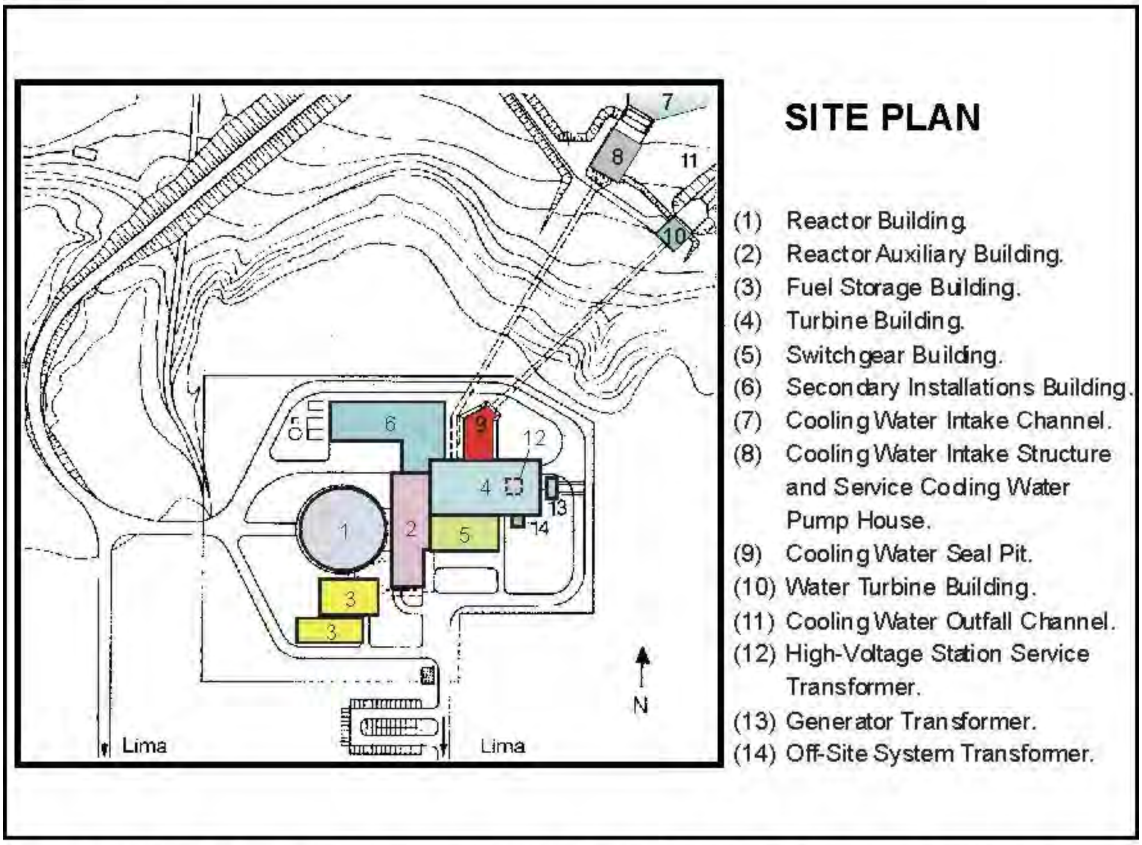
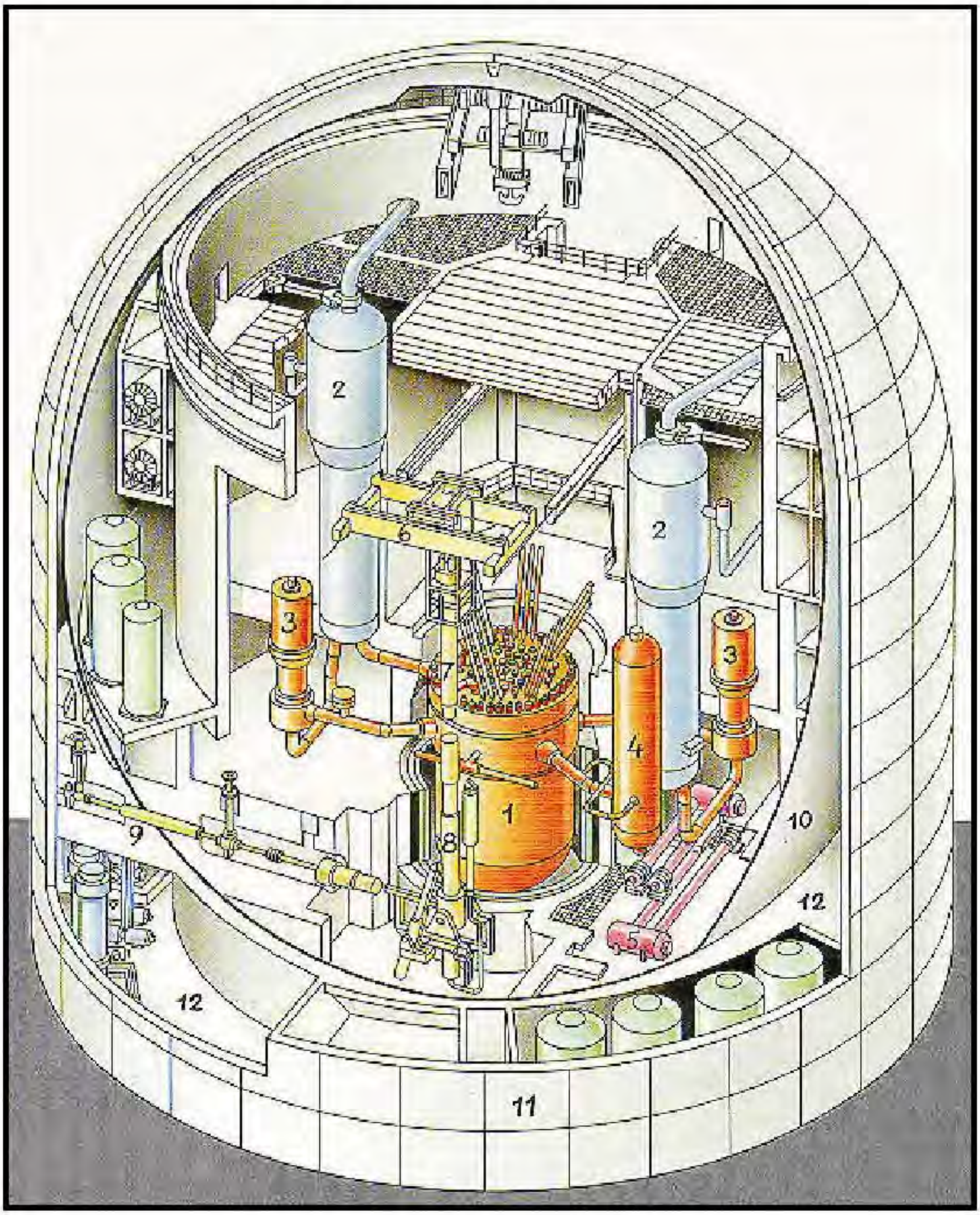


Figure V.1-3 - Atucha I Nuclear Power Plant - Main Buildings and Structures

REACTOR BUILDING



1 - Reactor pressure vessel
2 - Steam generator
3 - Reactor coolant pump
4 - Pressurizer
5 - Moderator cooler
6 - Refueling machine travelling gear
7 - Refueling machine
8 - Tilter
9 - Fuel transfer tube
10 - Containment
11 - Reactor Building.
12 - Annulus

Figure V.1-4 - Atucha I Nuclear Power Plant - Reactor Building

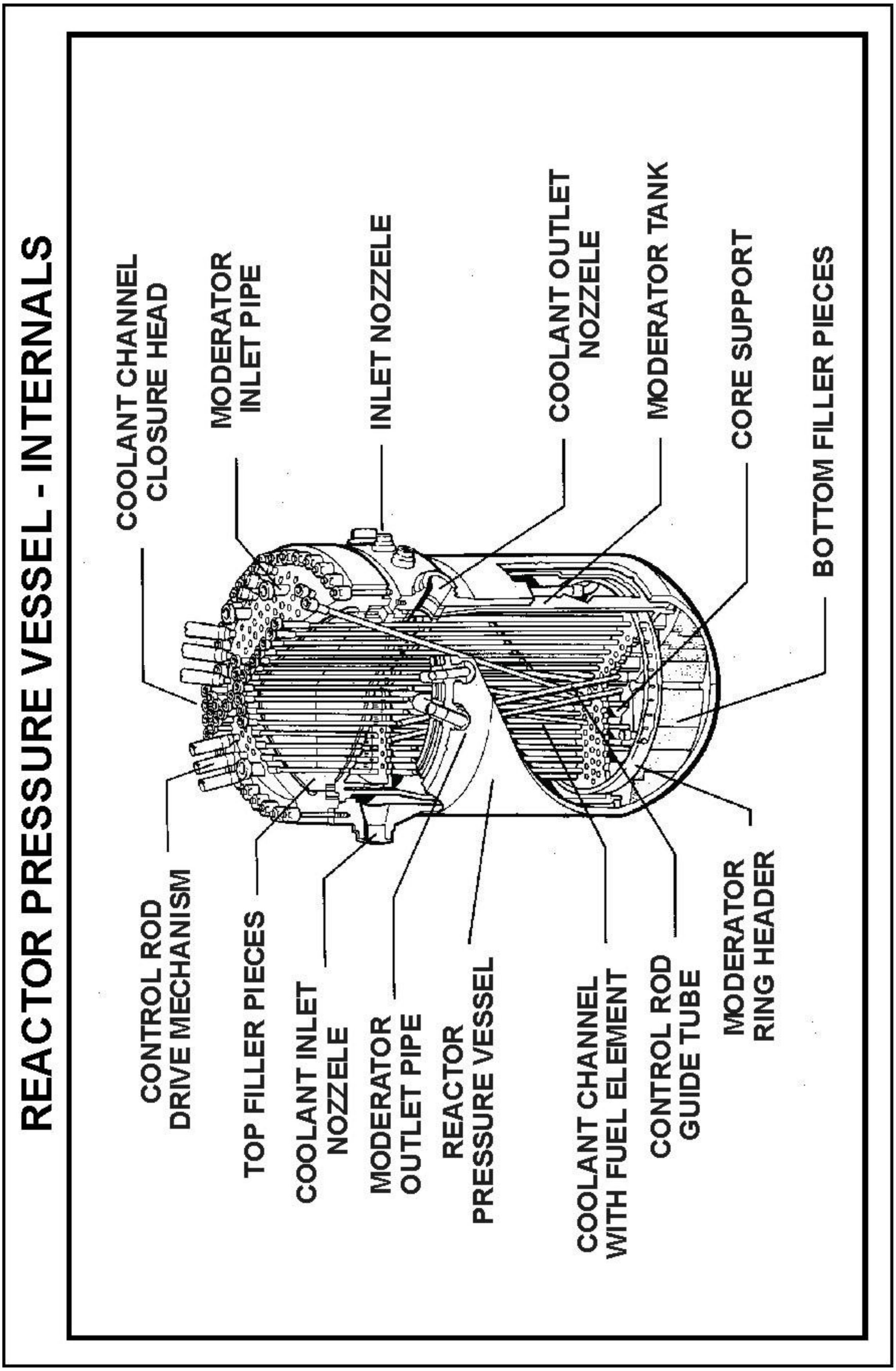
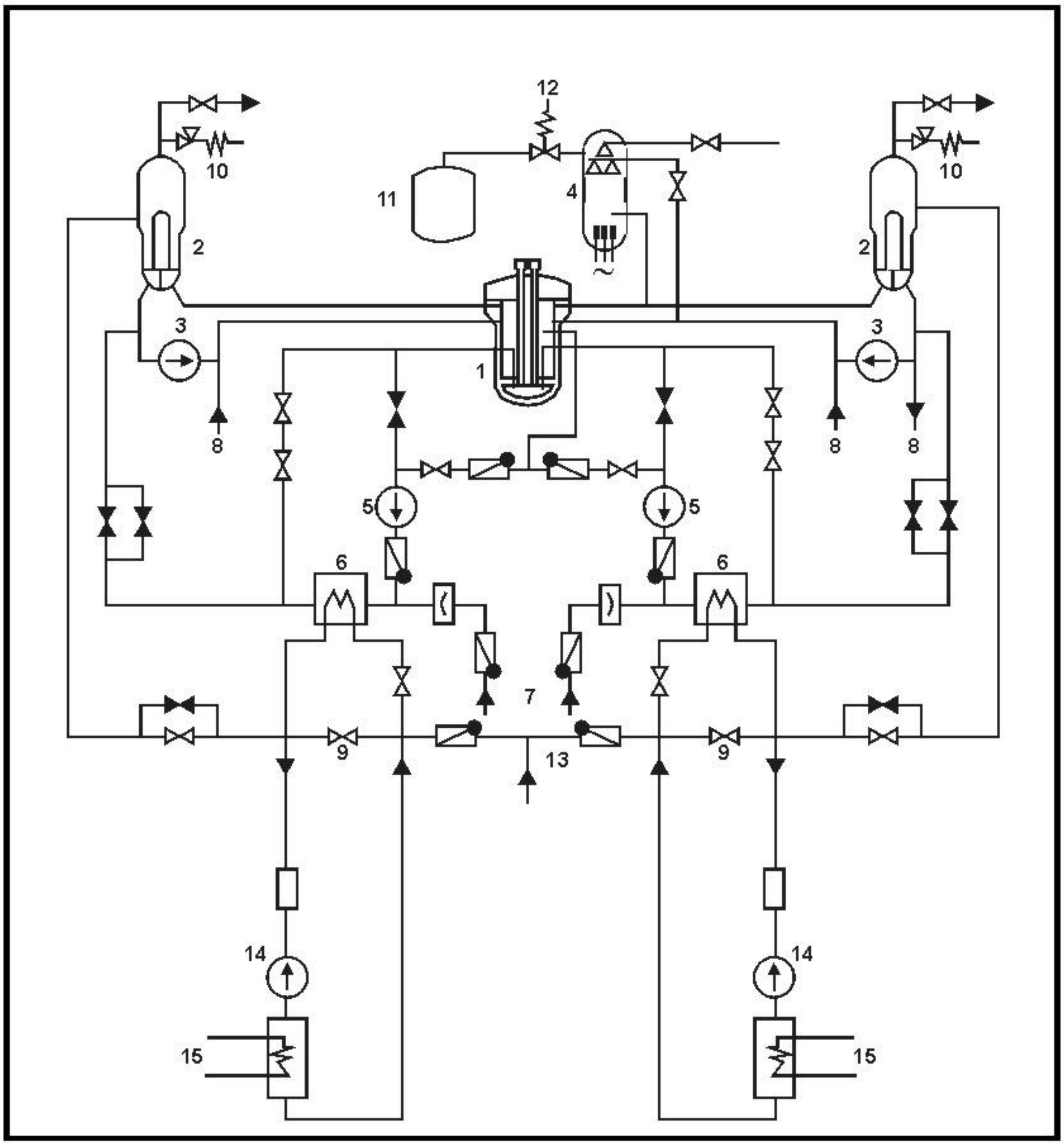


Figure V.1-5 - Atucha I Nudear Power Plant - Reactor Pressure Vessel and Internals

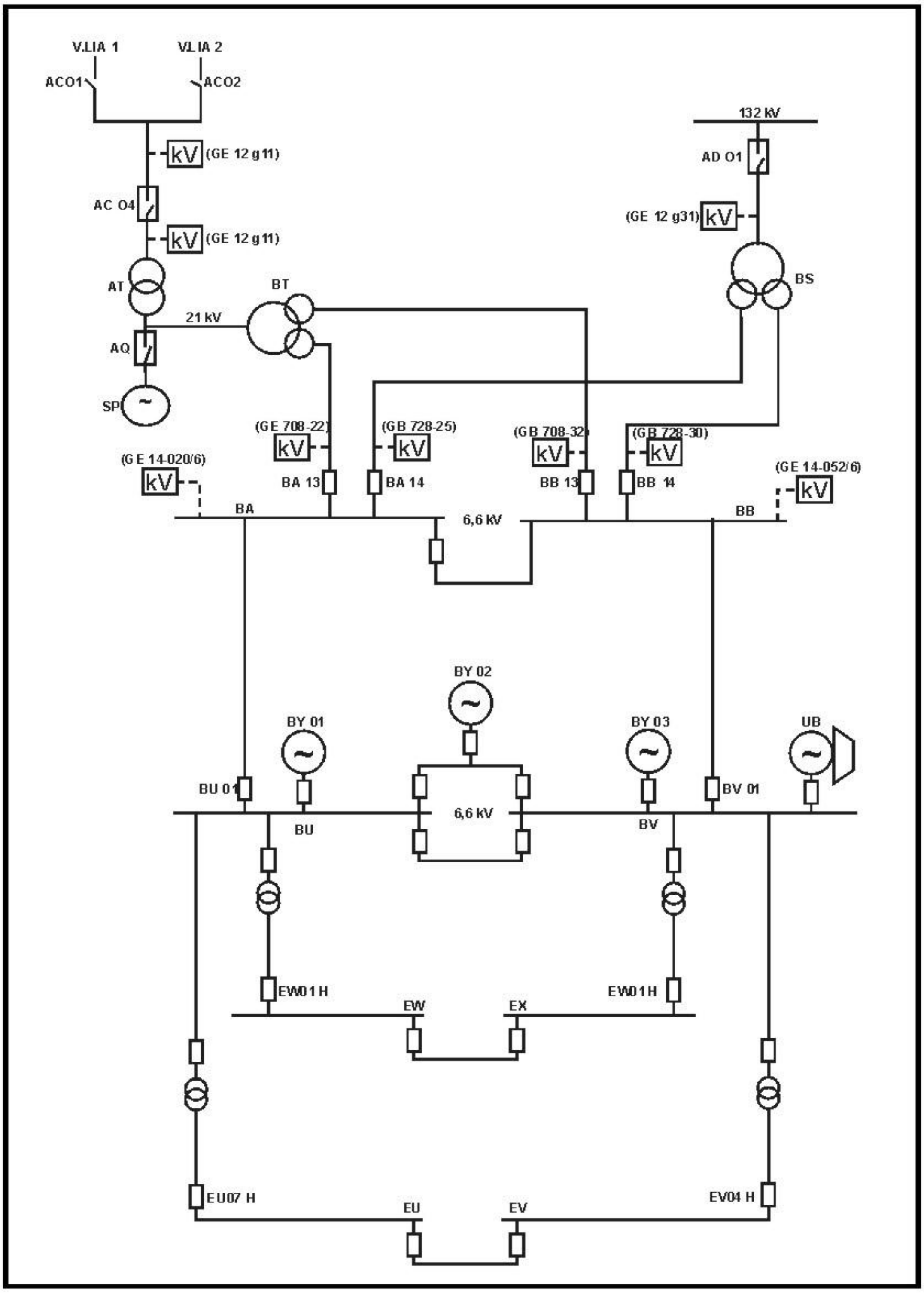
REACTOR COOLANT SYSTEM AND MODERATOR SYSTEM



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 V.1-6 - Atucha I Nuclear Power Plant - Reactor Coolant System and Moderator System

BASIC ELECTRICAL POWER SYSTEM (SINGLE LINE DIAGRAM)



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

Figure V.1-7 - Atucha I Nuclear Power Plant - Basic Electrical Power System

V.2. EMBALSE NUCLEAR POWER PLANT

V.2.1. INTRODUCTION

In 1967 the Comisión Nacional de Energía Atómica (CNEA) of Argentina initiated the feasibility study for the construction of Embalse Nuclear Power Plant (CNE) and in 1973 signed a contract with Atomic Energy of Canada Limited (AECL) and Societa Italiani Impianti P.A. (IT) for a 600 MWe CANDU-PHW (pressurized heavy water) type nuclear power plant at the Embalse site in the Province of Córdoba, Argentina, on the Almafuerte Peninsula just out from the south shore of Rio Tercero Lake, as shown in ***Figure V.2-1***.

The construction of the station began in May 1974 and the commercial operation in January 1984.

At present, the owner of CNE is Nucleoeléctrica Argentina SA, and the plant provides a net electric power of 600 MWe to the interconnected national system.

The plant is designed for commercial base-load operation. It contains a turbine generator set, with steam supply from a CANDU-PHW type nuclear reactor. This design has been used in all Canadian designed nuclear power plants built to date, with the exception of Gentilly-1.

Besides, the plant also has components, equipment and sub-systems required for the functioning of the big systems located at its "nuclear" and "conventional" sections.

The CANDU-PHW type reactor uses heavy water as moderator and as a heat transport medium. The fuel is natural uranium supplied in the form of bundles loaded into and removed from the reactor during "on power" operation. Its thermal power is 1987 MWt. A closed loop cooling circuit is provided to transfer the heat from the fuel and to produce light water steam in the steam generators. The turbine cycle is similar to that which has been used for other plants of this type.

Figure V.2-2 shows schematically the main systems of the Embalse Nuclear Power Plant.

V.2.2. OVERALL PLANT LAYOUT

Building and structure arrangements of CNE are shown in ***Figure V.2-3***.

CNE's main buildings and structures may be classified into nuclear steam plant and balance of plant. The nuclear steam plant includes the reactor building, service building, emergency water supply building, high-pressure emergency core cooling building, and their contents except for balance of plant equipment in the control room. The balance of plant includes all other buildings and their contents.

The reactor building (***Figure V.2-4***) houses the reactor, fuel handling systems, the heat transport system, including the steam generators, and the moderator system, together with their associated auxiliary and special safety systems.

The reactor building is divided into three major structural components: the containment structure, the internal structure, and the reactor vault structure.

The containment structure is the main component of the containment system. This structure is a pre-stressed concrete building comprising three structural components: a base slab approximately 1.74 m thick; a cylindrical wall approximately 41.5 m diameter with a minimum wall thickness of about 1.07 m, and a spherical segmental dome with a thickness at the crown of about 0.60 m.

Beneath the outer dome there is a second dome having an opening in the crow, which together with the perimeter wall forms a container to provide storage for 2170 m3 of water for dousing and emergency core cooling.

The internal structure is a reinforced concrete building dividing the reactor building into two areas as follows: the "accessible area” to which operating and maintenance personnel have access during normal plant operation, and the "inaccessible area" which is not accessible during plant operation, but to which access can be obtained after plant shutdown. The internal structure is separated from the containment structure. All those system and items of equipment to which access is routinely required for operation, servicing or maintenance, are housed in rooms within the accessible area. Outside of the accessible area, the remainder of the reactor building forms the inaccessible area containing the reactor and its vault, the heat transport and moderator system, the fuelling machine operating areas, steam generator room, and the areas for auxiliaries. Service cranes are provided as required in this area.

The reactor vault structure is a reinforced concrete, carbon steel-lined, light water-filled tank which contains and supports the calandria and end shields. Adequate shielding is provided by the concrete vault for access within the reactor building during plant operation. The vault is independent of other structural units within the reactor building.

The service building is a conventional reinforced concrete structure with concrete floors. It contains the following main facilities: control room, spent fuel transfer and storage facilities, and heavy water treatment and radioactive waste treatment facilities. It also contains conventional and nuclear service facilities such as stores, workshops, charge rooms, a decontamination centre and laboratories.

The turbine building, consisting of a turbine hall and the turbine auxiliary bay, has a reinforced concrete main structure. The turbine hall houses the turbine generator and some associated auxiliary equipment. Other auxiliary equipment and electrical power distribution equipment are contained in the turbine auxiliary bay.

The auxiliary bay is adjacent to and structurally independent from the service building which forms part of the plant. The main access leading to the loading bay In the turbine building is at the end of the turbine hall.

The building complex has reinforced concrete foundations and structures. The turbogenerator pedestal is a reinforced concrete structure rising from the foundations slab. Only the roof of the turbine building is structural steel work.

The other main structures of the station are: diesel building, emergency water supply pumphouse, and water supply structures.

The diesel generator and emergency steam generator building has reinforced concrete slabs. The diesel building consists of one single building formed of concrete walls and roof. This building is divided into five rooms, four of which contain generator sets together with the related control panels. The fifth room is used for storage purpose. Partition walls between these rooms are full height reinforced concrete. The auxiliary steam generators are located outdoors near the diesel building.

The emergency water supply pumphouse is a reinforced concrete structure with a floor elevation of 97.0 m. Two removable hatches in the concrete roof slab are located over the diesel-driven pumps.

The water required for the different services of the station is taken from the reservoir of Embalse by means of the water supply structures. These structures include: pump house, water intake structure, and the water circulation piping.

V.2.3. CNE MAIN SYSTEMS

In what follows the main safety and process systems that are part of the station, are briefly summarized.

V.2.3.1 Reactor

The Canadian heavy water-moderated, natural uranium-fuelled, pressurized heavy water reactors utilize the “pressure tube" concept. This consists of an array of pressure tubes, containing the reactor fuel, passing through a large cylindrical vessel (the calandria) containing the heavy water moderator and reflector.

Pressurized heavy water coolant is pumped through the pressure tubes, cooling the fuel and conveying heat from the fuel to the outlet header and to the steam generator. Each pressure tube is isolated and insulated from the heavy water moderator by a calandria tube. The annular space between concentric pressure and calandria tubes is filled with a gas.

It should be noted that this type of design results in a partially redundant structure, insofar as any localized failure of the moderator boundary will not result in the failure of the structure as a whole, and is therefore tolerable from a safety standpoint.

The reactor assembly ***(Figure V.2-5)*** comprises the calandria assembly within the calandria vault, fuel channel assemblies and reactivity control units. The calandria vault is an ordinary carbon steel-fined concrete structure, and is filled with light water. The water serves as a thermal shield and as a cooling medium.

The calandria assembly comprises the calandria, two end shields, and an embedment ring at each end shield (the embedment rings are grouted into the concrete wall of the calandria vault). This assembly forms an integral multi-compartment structure which provides containment for the heavy water moderator and reflector, the fuel channels (less end fittings), the reactivity control units, and the reactor shielding.

The calandria comprises a horizontal, cylindrical, single-walled, stepped shell, enclosed at each end by the tubesheet of an integral end shield, and spanned horizontally by 380 integral calandria tubes. The functions of the calandria are the following:

- contains the heavy water moderator and reflector enveloping the in-core portions of the fuel channels.
- helps support the in-core components of the reactivity control units.
- helps support the fuel channels.
- helps support the moderator piping, and any other piping, attached to it.

The calandria is designed for a postulated pressure tube/calandria tube rupture. To limit the pressure resulting from such an accident, four pressure relief pipes are provided. These pipes extend from the top of the calandria through the shield light water in the calandria vault, and terminate at the rupture discs located in the top of the calandria vault adjacent to the reactivity mechanism deck.

The end shields are horizontal, cylindrical shells enclosed at each end by tubesheets, and spanned horizontally by 384 lattice tubes. They contain biological shielding material in the form of carbon steel balls and ordinary light water. The functions of the end shields are as following:

- shields the fuelling machine areas from the reactor during reactor operation and during shutdown.
- helps support the calandria
- helps support and align the fuel channels
- provides a gas-filled annulus between the hot end fittings and lattice tubes in order to minimize the heat loss.

Two end shields are integral parts of the calandria assembly, one end shield being welded to each end of the calandria. Outside of each end shield, and concentric to it, is the end shield embedment ring which is grouted into the calandria vault wall. The end shield support structures are designed to accommodate the differential movements between the reactor assembly and the calandria vault which result from thermal and loading effects.

Each fuel channel assemblies consists of a zirconium-niobium alloy pressure tube expanded at each end into the hub of an alloy steel end fitting. Each assembly with its fuel and heavy water coolant is supported by the end shield lattice tubes through sliding bearings and, partially, by the calandria tube/pressure tube annular spacers. The end fittings are designed to allow relative axial movement between the fuel channel assemblies and the lattice tubes to cater for thermal expansion and pressure tube creep.

The inlet and outlet end fittings are designed to meet the following requirements:

- to provide a suitable high pressure closure that can be operated by the fuelling machine to allow insertion and removal of fuel.
- to provide shielding in the end shield penetrations to allow service access to the fuelling machine operating areas and to the face of the end shields at shutdown.
- to provide a transition between the pressure tubes and the primary circuit piping.
- to provide support for the pressure tubes and their contents.

The channel closure consists of a flexible seal disc mounted on a body witch locks firmly into the end fitting by means of a set of extendable jaws. The seal disc bears against a seal face in the end fitting to prevent leakage and is nickel plated to improve leak tightness. A shield plug is locked into each end fitting where the end fitting passes through the end shield. Both the channel closure and the shield plug can be removed and reinserted by the fuelling machine during refuelling.

The fuel is designed to be compatible with the operating conditions imposed on it by the heat transport system, the fuel handling system and reactor nuclear design.

The reactor is fuelled with natural uranium in the form of compacted and sintered cylindrical pellets of uranium dioxide. About thirty-five uranium dioxide pellets are stacked end-to-end and are sealed in a zirconium alloy sheath to form a fuel element. An interlayer of graphite between the pellet stack and the sheath is used to reduce the pellet/sheath interaction. These fuel elements are also fitted with zirconium alloy spacers and bearing pads. Thirty-seven fuel elements are welded to two zirconium alloy end plates to form the cylindrical bundle. The end plates maintain separation among the fuel elements at the bundle element extremities.

The separation among the fuel elements at the bundle mid-length is maintained by the spacers which are brazed to the fuel elements. The spacers are positioned on each individual fuel element such that the contact between any two mating elements is spacer-to-spacer. Bearing pads are brazed to the outer ring of fuel elements. The three planes of bearing pads maintain proper clearances between the bundle and the fuel channel during fuel handling operations and during the bundle's residence in the reactor.

Concerning reactivity control units it should be mentioned that neutron absorbing devices, both liquid and solid, are provided to control reactivity. During operation, reactivity is controlled by adjuster units, control absorber units, and zone control units. Under emergency or abnormal conditions, reactor shutdown is quickly achieved by dropping shutoff absorbers into the reactor core, or by injecting liquid poison into the heavy water moderator.

Twenty-one vertical adjuster units are provided, each comprising an assembly of zircaloy clad cobalt absorber elements, a vertical guide tube and a drive mechanism. The absorber shape the neutron flux for optimum reactor power and fuel burnup when inserted in the calandria, and upon removal from the calandria allow excess reactivity for overriding xenon poison following a power reduction.

Four control absorber, mounted vertically, adjust the flux level at times when greater reactivity rate or depth is required than that provided by the zone control system. The design is essentially the same as that of the shutoff units, except that the shutoff unit accelerator spring is omitted from the design.

The zone control units are tubular members divided into compartments within the reactor core, each capable of being filled to any desired level with light water. There are six vertically oriented zone control units in the reactor. The units are used to adjust the flux level in any one of fourteen zones in the reactor. This is accomplished by introducing a continuously controlled amount of light water into the zones to provide a local control of neutron absorption.

On the other hand the reactor has two shutdown systems: the shutoff units and the liquid poison injection system; these systems are discussed in section 3.9.

V.2.3.2 Heat transport system

The heath transport system circulates pressurized heavy water (reactor coolant) through the reactor fuel channels to remove heat produced by fission of uranium fuel. The heat is carried by the reactor coolant to the steam generators where it is transferred to light water to form steam, which subsequently drives the turbine generator.

The major components of the heat transport system are the reactor fuel channels, four vertical steam generators, four motor driven pumps, four reactor inlet headers, four reactor outlet headers, one electrically heated pressurizer, and all necessary interconnecting piping and valving. The fuel channels are horizontal and allow access to both ends by the fuelling machines. The header, steam generators and pumps are located above the reactor. The normal operation flowsheet for the heat transport system is show in ***Figure V.2-6***.

The main features of the transport system are as follows:

- circulation of the reactor coolant is maintained at all times during reactor operation, shutdown and maintenance.
- each heat transport pump has sufficient inertia in rotating components to prevent a sudden decrease in the flow if power to the pump motor is lost.
- adequate heat transport system flow for shutdown heat removal is maintained by natural convection flow following pump rundown.
- heat transport system pressure is controlled at an acceptable value for all normal modes of operation.
- system components are protected from overpressure by instrumented relief valves and suitable reactor regulating and/or safety system action.
- a separate shutdown cooling system is provided to remove reactor shutdown heat, thus permitting the draining of steam generators and pumps in the heat transport system, for maintenance.
- purification by filtering, ion exchange and degassing is provided to control the chemistry of the reactor coolant.
- potential heavy water leak sources are kept to a minimum by using welded construction and bellows sealed valves wherever practicable. Where potential leak sources exist, they are connected to closed collection and recovery systems.

The heat transport system has two loops in order to reduce the rate of positive coolant void reactivity insertion in the event of a loss-of-coolant accident. The two loops each contain two steam generators, two pumps, two reactor inlet headers, two reactor outlet headers, one set of inlet feeders and one set of outlet feeders. Feeders flow are matched to individual fuel channel powers to give an equal heavy water steam quality for each channel at the reactor outlet headers when the reactor is at full power. Pressure drop causes the heavy water steam quality to increase at the inlet to the steam generator.

The two figure-of-eight loops provide bi-directional flow through the core such that the flow is in opposite directions in adjacent channels. Each loop removes the heat from half of the fuel channels in the reactor core. Each loop has one inlet and outlet header at each end of the reactor core. Heavy water is fed to each of the fuel channel through individual feeder pipes from the horizontal reactor inlet headers, and heavy water is returned from each fuel channel through individual outlet feeder pipes to the horizontal reactor outlet headers. Individual feeder piping sizes depend on the coolant flow to the particular channel.

The pressure in the reactor outlet headers is controlled by a common pressurizer connected to a line linking the outlet headers at one end of the reactor. Valves in these lines provide isolation between the two loops in the event of a loss-of-coolant accident.

Two pipes connect each reactor outlet header to one steam generator. As the reactor coolant passes through the four steam generators, heat is removed and the reactor coolant at the outlet of the steam generator is sub-cooled. Each steam generator is connected to the pump suction of one heat transport pump by one pipe, and each heat transport pump delivers coolant to one reactor inlet header through two pipes.

V.2.3.3 Moderator system

Neutrons produced by nuclear fission are moderated by the heavy water in the calandria. The heavy water is circulated through the moderator system of cooling, for purification and for control of the concentration of substances used for reactivity adjustment. ***Figure V.2-7*** is a simplified flow diagram for the moderator system. The system consists basically of two 100 % capacity pumps (connected in parallel) which are connected in series with two 50 % capacity heat exchangers (connected in parallel). The series/parallel arrangement permits the operation of either pump with the two heat exchanger. Main moderator system connections are provided for the purification, liquid poison addition, heavy water collection, heavy water supply and heavy water sampling systems.

The heavy water in the calandria functions as a heat sink in the unlikely event of a loss-of-coolant accident coincident with failure of emergency core cooling. The capacity of the heat sink is assured by controlling the heavy water temperature in the calandria at a constant value.

Potential heavy water leak sources are kept to a minimum by using welded construction, seal welding, and bellows seals wherever practical. Where potential leaks sources do exist in the moderator system, the leak sources are connected to the heavy water collection system. The reliability of the moderator system is assured by appropriate component, Instrument and power supply redundancies.

The main moderator system pumps, valves and heat exchangers are in compact arrangement at approximately grade elevation to one side of the calandria vault. The pump suction lines and heat exchanger outlet lines are anchored to a rigid penetration seal where they pass through the calandria vault concrete to eliminate any possibility of loss of vault shielding water.

The moderator pump motors are connected to the high voltage Class III power supply. In addition, each pump has a pony motor capable of driving the pump at 25 % speed and connected to the low voltage Class III power supply. In the event of a loss of Class IV power the power to the main motors is lost until the diesel generators can supply Class Ill power. The cooling water supply to the heat exchanger is also re-established after three minutes at a lower flow following a total failure of Class IV power. The rate of heat removal is sufficient to limit the increase of moderator temperature in the calandria to an acceptable value during a failure of Class IV power and subsequent reactor shutdown.

The heavy water in the calandria is maintained at relatively uniform temperature and circulated to eliminate hot spots. The circulation is promoted by pumping the heavy water from the bottom of the calandria and, after cooling, returning it through nozzle jets inside both sides of the calandria at the horizontal diameter.

Live-loaded double-packed stem seals are used on large valves in the moderator system to reduce leakage and maintenance. Bellows stem seals are used on small valves. All of the equipment in the moderator system is accessible for isolation and maintenance when the reactor is shutdown. Space for heat exchanger shell removal and other provisions for maintenance are features of the equipment arrangement.

V.2.3.4 Fuel handling system

The fuel handling system comprises equipment for storage of new fuel for fuel changing and for temporary storage of spent fuel. Reactor fuel is changed on a routine basis with the reactor operating at full power. Space and lifting facilities are provided for shipping spent fuel. The new fuel storage room, the fuelling machine decontamination and service rooms and the spent fuel storage rooms are located in the service building. The fuelling machines, which load and unload the fuel discharge equipment are normally operated remotely and automatically from the control room of the plant. Personnel are only required to enter the reactor building to load new fuel into the new fuel transfer mechanism and for maintenance of the fuel handling system components. These access areas are provided with full biological shielding.

Storage and handling facilities are provided to accommodate bulk storage of fuel in the service building, safe transfer of fuel to the reactor building and easy manual loading of new fuel bundles into the motorized new fuel ports. New fuel is received in packages in the new fuel room in the service building. This room can accommodate 9 month's fuel inventory and can store temporarily the fuel for the initial loading. When required, the packages with new fuel are transferred to the new fuel loading area in the reactor building. Here the bundles are identified, inspected and loaded manually into the magazines of the two fuel ports which penetrate into the fuelling machine maintenance locks. Mechanisms of the ports are motorized and can be controlled remotely. To load a fuelling machine with new fuel, the machine locks on to the port and normally accepts up to 10 bundles into a magazine within the head. The spent fuel discharge and storage equipment is sized for the accumulation of fuel over a period of 10 years at the average fuelling rate of the reactor, with provision for loading a shipping flask with fuel underwater.

After the minimum decay period established in 6 years, spent fuel elements are transferred to special dry storage silos, also located inside the nuclear power plant site. The fuel elements are introduced in stainless steel baskets, each of them containing up to 60 fuel elements vertically arranged in a circular grid; this operation is carried out under water. Later on, the baskets are sent to the transfer building, where the lid is weld. Finally they are introduced in a special container providing enough shield and containment (transfer 'flask") to be transported to the silo field where they are stored. Each silo contains 9 baskets.

Handling and storage space is also provided underwater for irradiated parts of the reactivity control mechanisms which may be removed from the reactor, and for shipping irradiated cobalt adjuster bundles from the station.

The spent fuel handling system consists of: discharge and transfer in the reactor building, reception, storage bays in the service building, and dry storage system. The transfer of spent fuel between buildings is under water through a transfer channel. The discharge and transfer operations are controlled remotely, while operations in the storage bays are carried out manually underwater using long tools, and aided by powered cranes and hoists. The equipment incorporates devices for canning failed fuel bundles and is arranged to reduce the radiation exposure of personnel when handling failed fuel to acceptably low levels. The discharge equipment comprises two valved spent fuel ports located above the water level, while the transfer equipment is located in a shielded room and extends down under the water and into a reception bay in the service building.

The on-power fuel changing equipment is located in the reactor building and consists of two identical, unshielded fuelling machines, which are operated remotely. The fuelling machines are normally stored In two fuelling machine maintenance locks and are suspended by tracks. Each set of tracks connects with a bridge at each face of the reactor. Powered shielding doors separate the maintenance locks from the reactor and, when closed, allow access to the fuelling machines while the reactor is at full power. While in the maintenance locks the fuelling machines can lock on to the new fuel port to accept new fuel, to the service port for maintenance or service, or on to the spent fuel port discharge spent fuel.

The fuel loading is based on the combined use of the two remotely controlled fuelling machines operating at each end of a fuel channel. New fuel bundles, from one fuelling machine, are inserted into a fuel channel in the same direction as the coolant flow and the displaced irradiated fuel bundles are received into the second fuelling machine at the other end of the fuel channel. Either machine can load or receive fuel. The direction of loading depends upon the direction of coolant flow in the fuel channel being fuelled, which alternates from channel to channel. The fuelling machine receive new fuel while connected to the new fuel port and discharge irradiated fuel while connected to the discharge port. The entire operation is directed from the control room.

V.2.3.5 Auxiliary systems

The main auxiliary systems of the Embalse Nuclear Power Plant are the heat transport auxiliary systems and the moderator auxiliary systems.

The heat transport auxiliary systems are the following:

- heat transport system purification circuit: this system minimizes the accumulation of radioactive corrosion products in the circuit, controls the fission products concentration (iodine) released by defective fuel elements, and contributes to a proper control of the coolant pD.
- gland seal system: it supplies filtered heavy water at high pressure to the heat transport pump glands.
- shutdown cooling system: it cools the heat transport system from 170 °C down to 54 °C and holds the system at that temperature indefinitely.
- heat transport pressure and inventory control system: it provides the pressure and inventory control for each heat transport circuit, and provides overpressure protection and a controlled degassing flow.
- heat transport heavy water collection system: it collects leakage from mechanical components, and receives heavy water sampling flow, and heavy water drained from equipment prior to maintenance.
- heat transport heavy water sampling system: it is used to obtain samples of heavy water from various points in the heat transport system. The samples are tested in the laboratory for pD, conductivity, chloride, tritium, lithium, dissolved gases, fission products and corrosion products.
- steam and feedwater systems: they enable the live steam supply to the plant turbine generator, the control of the feeding water level and the vapour pressure in the steam generators, the steam release to the atmosphere under certain situations of the station, and an adequate protection against overpressures in the steam generator secondary circuit.

The moderator auxiliary systems are the following:

- moderator purification system: it maintains the heavy water purity, thereby minimising radiolysis which may cause excessive build-up of deuterium in the cover gas; minimises corrosion of components and crud activation by removing impurities present in the heavy water and by controlling the pD; removes soluble poisons, boron and gadolinium, used for reactivity control in response to reactivity demands; removes the gadolinium, after initiation of the liquid injection shutdown system.
- moderator cover gas system: it prevents the accumulation of gaseous deuterium and oxygen produced by water radiolysis of the moderator in the calandria. The system recombines deuterium and oxygen catalytically, generating heavy water. The cover gas used in the moderator system is helium, because it is an inert gas and is not activated by neutron irradiation.
- liquid poison system: this system adds negative reactivity to the moderator to allow for excess reactivity in new fuel; adds negative reactivity to the moderator to allow for loss of xenon reactivity after a poison-out or long shutdown; provides a means of decreasing reactivity together with other reactivity control devices; provides a means to guarantee enough poison in the moderator to prevent criticality during shutdown.
- moderator heavy water collection system: this system collects heavy water leakage from the moderator pump seals, from the interpacking space of the main moderator gate valves, and from the intergasket of the main moderator heat exchangers.

V.2.3.6 Control centre

The control centre is a clean air conditioned area comprising the main control roam and adjacent auxiliary control areas. The control centre is divided into four basic areas. The centre portion contains the main control panels, the operator's desk and the fuelling machine and fuel handling console. A control equipment area containing the bulk of the control and instrumentation equipment for the unit is located behind the main control panels. The plant computers are located in a room behind the switchyard and electrical services panel. Also located in the control centre are a shift supervisor's office, a technical office, a washroom and a work control area.

The control room contains a unit control panel, an electrical services panel, an operator's desk with two high speed line printers, and a fuel handling and fuelling machine control console. The control room instrumentation is based on the philosophy of having sufficient information displayed to allow the unit to be controlled safely from the control room. To achieve this goal, all indications and controls essential for operation (start-up, shutdown and normal) are located on the control room panels. Also located there are controls for any systems requiring attention within 15 minutes of an alarm occurrences. For system no requiring attention within 15minutes, local control may be provided.

Most information is presented to the operator via the station computer system. However, sufficient conventional display, annunciation and recording of plant variables is included to allow the plant to be properly run in the shutdown condition with both computers out of service.

In case the control room becomes uninhabitable, enough display and control instrumentation is provided at a location remote from the control room (the secondary control area) to allow the plant to be shut dawn and maintained in a safe shutdown condition.

V.2.3.7 Instrumentation and control systems

The instrumentation in the CNE encompasses a variety of equipment, designed to perform a number of monitoring, control and display functions. Nuclear instrumentation is provided to allow automatic control of reactor power and flux shape and to monitor local core behaviour. Conventional instrumentation provides signal for control and display of other plant variables. Central to the instrumentation and control system is a dual digital computer system. The plant is automated to require a minimum of operator actions during all phases of operation. All major control loops use the two computers as direct digital controllers, giving a redundant and highly reliable system which is powerful and flexible. Conventional analog control instrumentation is used on smaller local loops.

Instrumentation and control systems are designed to a large variety of detailed requirements, depending on their function, importance and physical environment. However, all the systems are designed according to the following general criteria:

- The maximum practical amount of automatic control is incorporated in the design, to allow CNE to be operated safely with a minimum staff, and to leave operating staff free for high level monitoring of overall plant status.
- Adequate redundant information is designed to be available to the operator at all times, in order to allow him to assess the status of the plant quickly, and to intervene with manual actions if necessary.
- Equipment is designed for a minimum of regular maintenance. Any necessary maintenance operations are kept as simple and speedy as possible.
- The instrumentation and control systems are designed for a high reliability and availability, to maximize both plant availability and safety. This reliability is achieved through a combination of component selection and design and through redundancy.
- The control systems are designed to make CNE tolerant to expected and unexpected transients in order to prevent unnecessary plant outages.
- Where possible, the control system are designed to prevent or minimize damage to plant equipment.
- The control systems are designed to minimize the number of unnecessary reactor trips by strong control action. A design objective is to make the intervention of the shutdown systems unnecessary (although not necessarily avoidable) in all cases except real accidents in which public safety is threatened.
- The instrumentation and control design complies with the applicable standards.

V.2.3.8 Electric power system

The Embalse Nuclear Power Plant has two physical independent grid connections (***Figure V.2-8***). One of them is the 500 kV grid and the other is the 132 kW grid (offsite power system). The generator supplies power to the 500 kV grid through the main output transformer. During normal station operation, the auxiliary service is supplied through the unit service transformers, which are supplied by the generator output. On start-up, the station auxiliary service power supply is provided by the system service transformers, which are supplied from the electrical grid via the switchyard at 132 kV.

A load switch is provided on the 22 kV generator bus bars. The purpose of the load switch is to make possible station start-up having the auxiliary services supplied from the 500/22 kV main transformer and from the 22/6.6 kV transformers as alternative to the 132/6.6 kV transformers. In this eventuality the parallel between the 500 kV grid and the generator is achieved using the load switch.

To provide electrical power with higher than usual reliability to the Class IV and Class III loads, an automatic transfer scheme is incorporated which ensures continuity of supply in the event of a failure of the unit or a failure of the system supply. Standby diesel generators and station batteries are provided.

The electrical system of CNE is similar to that found in conventional large thermal stations, with modifications introduced to satisfy the increased redundancy requirements. This results in a more selective bus arrangement and more standby and redundant equipment.

The station service system is designed to meet the following major design criteria from safety and reliability point of view:

- following a fault resulting in the severance of the unit from the grid, the unit must be able to supply its own station services.
- dual bus or better reliability must be provided.
- the system must be stable under fault conditions.
- the design must meet the requirements of all classes of power and lend itself to automatic and emergency transfer schemes.
- simplicity and economy are to be maintained.

The safety and reliability requirements of the CNE onsite power system are realized by applying two main concepts:

- the subdivision of power according to reliability of supply (classes of power and emergency power supplies to safety related systems)
- the odd and even supply concept which relates to redundancy of supplies and loads.

As regards the subdivision of power according to reliability of supply, it should be mentioned that:

1) The CNE service system buses are classified in order of their four levels of reliability to provide power during the routine operating states of the plant. The lowest number classified buses are the most reliable. These are as follows:

- Class IV power supply: normal ac supplies to auxiliaries which can tolerate long duration interruptions without affecting personnel or equipment safety. Complete loss of Class IV power initiates a reactor shutdown. Class IV power is the normal source of power to Class I, II, and III systems. The voltages for the supply Class IV are as follows: 6.6 kV ac – 380 V ac - 50Hz.
- Class III power supply: ac supplies to essential auxiliaries which can tolerate the short interruption required to start up and load the on-site standby generators, after the interruption of the normal (Class IV) supply sources. These essential auxiliaries are necessary for an orderly safe shutdown of the reactor. The voltages for the supply Class 111 are as follows. 6.6 kV ac – 380 V ac - 50 Hz
- Class II power supply: ac supplies for safety related and other essential loads. Power is normally provided through dc/ac inverter systems from the Class 1 dc buses. In the event of inverter system trouble, alternative power is supplied automatically from the Class Ill buses via appropriate transformers. Upon interruption of the normal power source (Class Ill) to the Class I bus the on-site battery supplies power without break until Class III is restored and hence the supply is termed "uninterruptable". The voltages for the supply Class II are as follows: 380 V ac – 220 V ac -- 50 Hz
- Class I power supply: dc supplies for safety related and other essential loads. Power is provided from on-site batteries when the normal power source (Class III via rectifier) is interrupted. The transfer of power supply from one source to another is without break and hence the supply is termed "uninterruptable". The on-site batteries are continuously charged from Class III sources. The voltages for the supply Class 1 are as follows: 220 V dc - 48 V dc

2) The standby power for the Class Ill loads is supplied by four diesel generator sets. These are housed in four separate rooms with fire resistant walls. Two diesel generators together are sized to supply the total shutdown of the unit with the exception of Class IV loads. The Class Ill shutdown loads are duplicated, one complete system being fed from two diesel generator sets. On loss of Class IV power the four diesel generators are required to start automatically. When they come up to speed and voltage, an automatic sequencing system will connect all necessary safety-related loads in a few seconds.

3) CNE emergency power supply system is provided as an independent backup electrical supply for certain safety related loads. It supplies power to facilitate safe shutdown of the reactor and decay heat removal in the event that the Class I, II, III, and IV power systems are unavailable or the main control room is uninhabitable due to a design basis earthquake. The system is seismically qualified and is also able to supply power to emergency core cooling valves to ensure that the emergency water supply system can supply makeup to the heat transport system after an earthquake. The voltages supplied by the emergency power supply system are as follows: 380 V ac- 220 V ac - 50 Hz and 48 V dc.

As regards the basic aspects of the odd and even supply concept, it should be mentioned that:

- the distribution systems for all classes of power at all voltage levels are divided into odd and even buses so that the dual bus, or better, reliability is provided.
- loads and redundant auxiliaries are connected wherever practical such that half of any process is supplied from an odd bus, and the other half from an even bus.
- auxiliaries supplied at a lower voltage than the associated primary element are connected to an odd or even bus to match the source for the primary element.
- the odd and even concept is also applied to the cable tray system, junction boxes, etc. in order to maintain physical separation between the odd and even systems.

V.2.3.9 Safety systems

Safety related systems are incorporated in the plant design to perform the following functions:

- shutdown the reactor and maintain it shut down
- remove decay heat and thus prevent subsequent process failures which might lead to accidental releases of radioactivity to the public
- supply necessary information for post accident monitoring to permit the operator to assess the state of the nuclear steam supply system
- maintain a barrier to limit the release of radioactive material to the environment

The systems included under the general term "safety related systems" are classified as special safety systems and safety support systems.

The special safety systems are incorporated in the plant to limit radioactive releases to the public for two classes of events: the single failure of a process system, and the single failure of a process system combined with the coincident unavailability of one of the special safety systems (a dual failure).

The CNE contains the following special safety systems:

- shutdown system N° 1 (shutoff units)
- shutdown system N° 2 (liquid poison injection)
- containment system
- emergency core cooling system

These systems are independent in design and operation and free from operational connection with any of the process systems, including the reactor regulating system, to the greatest possible extent.

The purpose of shutdown system N° 1 is to rapidly and automatically terminate reactor operation under emergency conditions. Twenty-eight vertical shutoff units are provided, each comprising a stainless steel sheathed cadmium absorber, vertical guide tube, and a drive mechanism. The system shuts down the reactor by releasing the cadmium absorber elements of the shutoff units, introducing negative reactivity. This release is initiated when any two of the three independent trip channels are actuated. When a reactor trip occurs, the reactivity control units of the regulating system automatically take a safe attitude. Typically, the liquid zone control compartments are flooded, the control absorbers are dropped, and the adjuster drives are shut off (the adjusters remain inserted or as is).

The purpose of shutdown system N° 2 is to rapidly and automatically terminate reactor operation independently of shutdown system N° 1. The system trips the reactor by injecting liquid poison into the bulk moderator when any two of the three independent trip channels are actuated. The system comprises injection nozzles, thimbles, bellows assemblies, gadolinium pressure vessels (poison tanks) ,a helium supply tank, a poison mixing tank, valves and piping.

The containment system is an envelope around the "nuclear" components of the heat transport system where failure of these components could result in the release of a significant amount of radioactivity to the public. Because of the large amount of energy stored in the heat transport system, the envelope must withstand a pressure rise. The criterion for determining the effectiveness of the envelope is the integrated leak rate for the period of the pressure excursion. To meet the design leakage requirements two approaches are taken. The first involves the detailed design of the envelope to minimize the leak rate. The envelope comprises a primary containment, and systems to filter and monitor the gas removed from the primary containment after a loss-of-coolant accident following dousing. The second approach involves the addition of a system that will absorb the energy released to the envelope, thus reducing the peak pressure and the duration of the pressure excursion. This energy absorbing system is composed of a source of dousing water, spray headers and initiating valves, and building air coolers.

The emergency core cooling system has three stages of operation: high, medium and low pressure. System operation is triggered, on a loss of coolant accident (LOCA), when the heat transport system pressure drops to 55.25 Kg/cm^2 and a circuit isolation system (independent of emergency core cooling system logic) closes the applicable valves to isolate the ruptured circuit.

The safety support systems provide reliable services, such as power and water, to the special safety systems, but may also perform other normal process functions in addition to their safety support roles. Because of the reliance on these systems for both normal plant operation and continuing operation of the special safety systems, special measures are taken in their design to assure reliability.

Two of the CNE safety support systems are the emergency water supply system and the emergency power supply system.

The emergency water supply system ensures that there is always sufficient water available to establish an adequate heat sink for decay heat removal when the normal source of such water is not available. The emergency power supply system is designed to act as an alternative source of electrical power for certain safety related loads when the normal source of supply is unavailable; this system was discussed in section 3.8.

V.2.3.10 Technical data of Embalse Nuclear Power Plant

Some of the main technical data are detailed in what follows:

Overall Plant Data	
Reactor type	CAN DU-PHW horizontal pressure tube. Model: CANDU 6
Net nominal electric power	600 MWe
Bulk nominal electric power	648 MWe
Authorised thermal power	2015 MWt

Reactor Core Data	
Type of fuel	Natural uranium
Shape of fuel bundle assembly	37 - rod cluster
Length of fuel bundle assembly	495 mm
Number of fuel channels	380
Cladding material	Zircaloy 4
Fuel bundles per channel	12
Refuelling	On load
Coolant and moderator	Heavy water

Primary Heat Transport System Data	
Pressure in the reactor inlet header	11.24 MPa
Pressure in the reactor outlet header	9.99 MPa
Temperature in the reactor inlet header	268°C
Temperature in the reactor outlet header	310 °C
Primary coolant flow	32.750 t/h
Heavy water concentration	More than 99.75 % (weight)

Turboset Data	
Stages	1 high pressure ; 3 low pressure
Speed outlet	1500 rpm
Steam pressure	46,2 Kg/cm^2
Steam flow	3.366 t/h
Condenser coolant flow	163,800 m^3/h
Generator type	Direct coupled, three-phase, four poles, hydrogen/water cooled
Generator power factor	0.85
Generator voltage output	22 kV
Generator frequency	50Hz

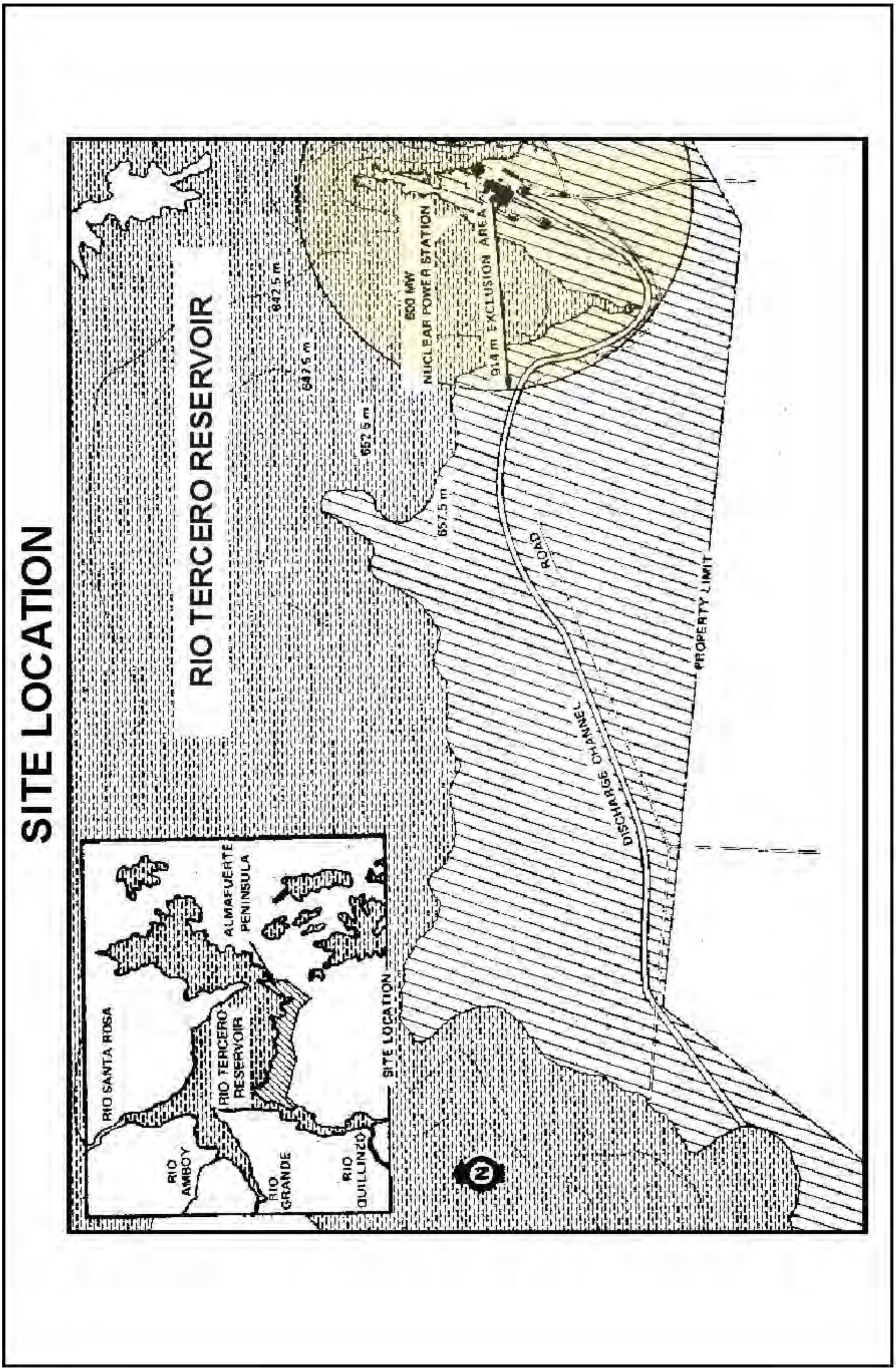


Figure V.2-1 - Embalse Nuclear Power Plant - Site Location

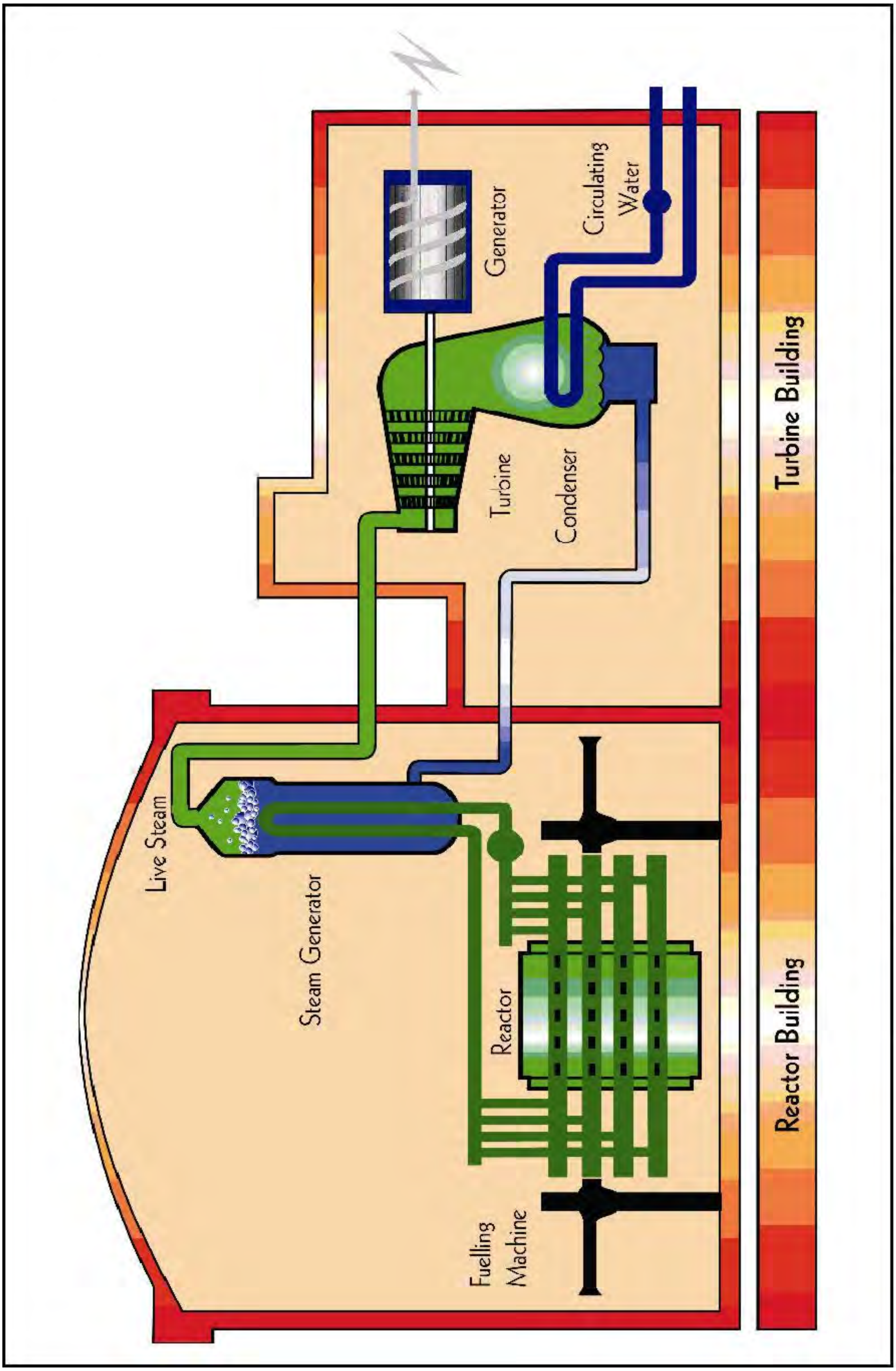
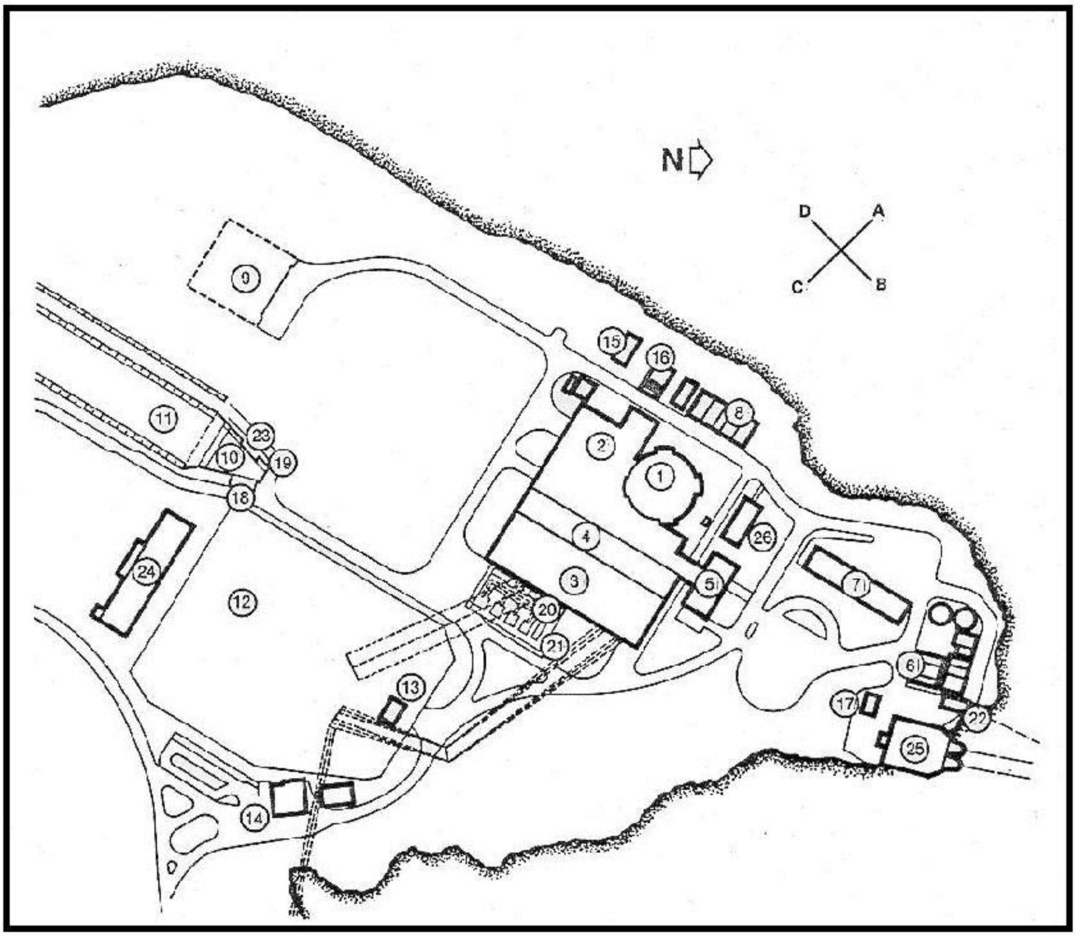


Figure V.2-2 - Embalse Nuclear Power Plant - Simplified Flow Diagram

SITE PLAN



1 REACTOR BUILDING
2 SERVICE BUILDING
3 TURBINE BUILDING
4 AUXILIARY BAY
5 ADMINISTRATION BUILDING
6 WATER TREATMENT PLANT
7 GARAGE
8 STANDBY GENERATOR
9 SOLID WASTE STORAGE
10 DISCHARGE WEIR
11 DISCHARGE CHANNEL
12 SWITCHYARD
13 COMMAND STATION
14 MAIN GATE
15 AUXILIARY BOILER
16 FUEL TANKS
17 SELF CLEANING FILTER
18 FIRE FIGHTING PUMPS
19 DRAIN PUMPS
20 TRANSFORMER AREA
21 HYDROGEN STORAGE
22 EMERGENCY WATER SYSTEM PUMP HOUSE
23 PROCESS WATER POOL
24 GENERAL WAREHOUSE
25 PUMPHOUSE
26 HIGH PRESSURE EMERGENCY CORE COOLING BUILDING

Figure V.2-3 - Embalse Nuclear Power Plant - Site Plan

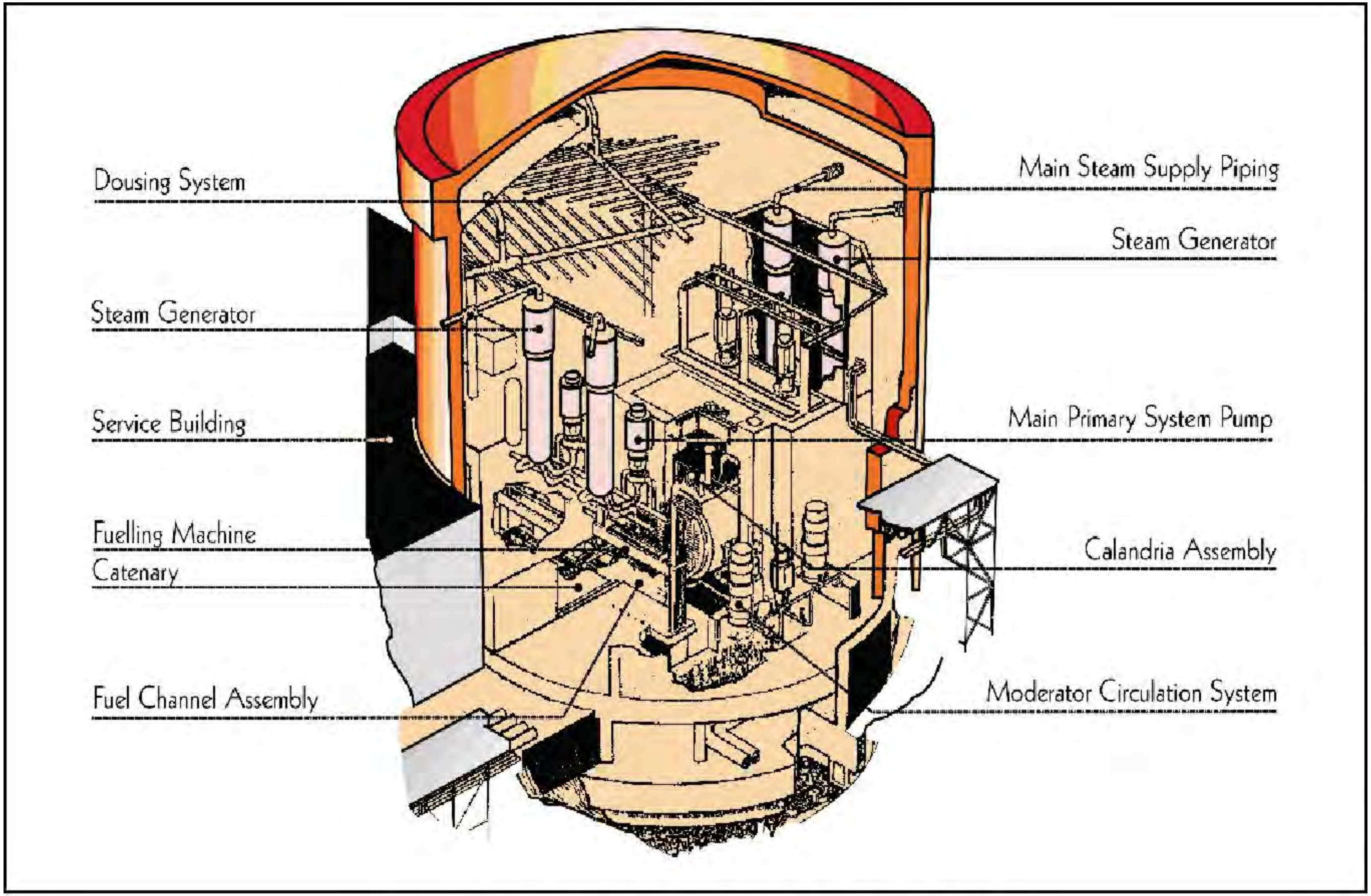
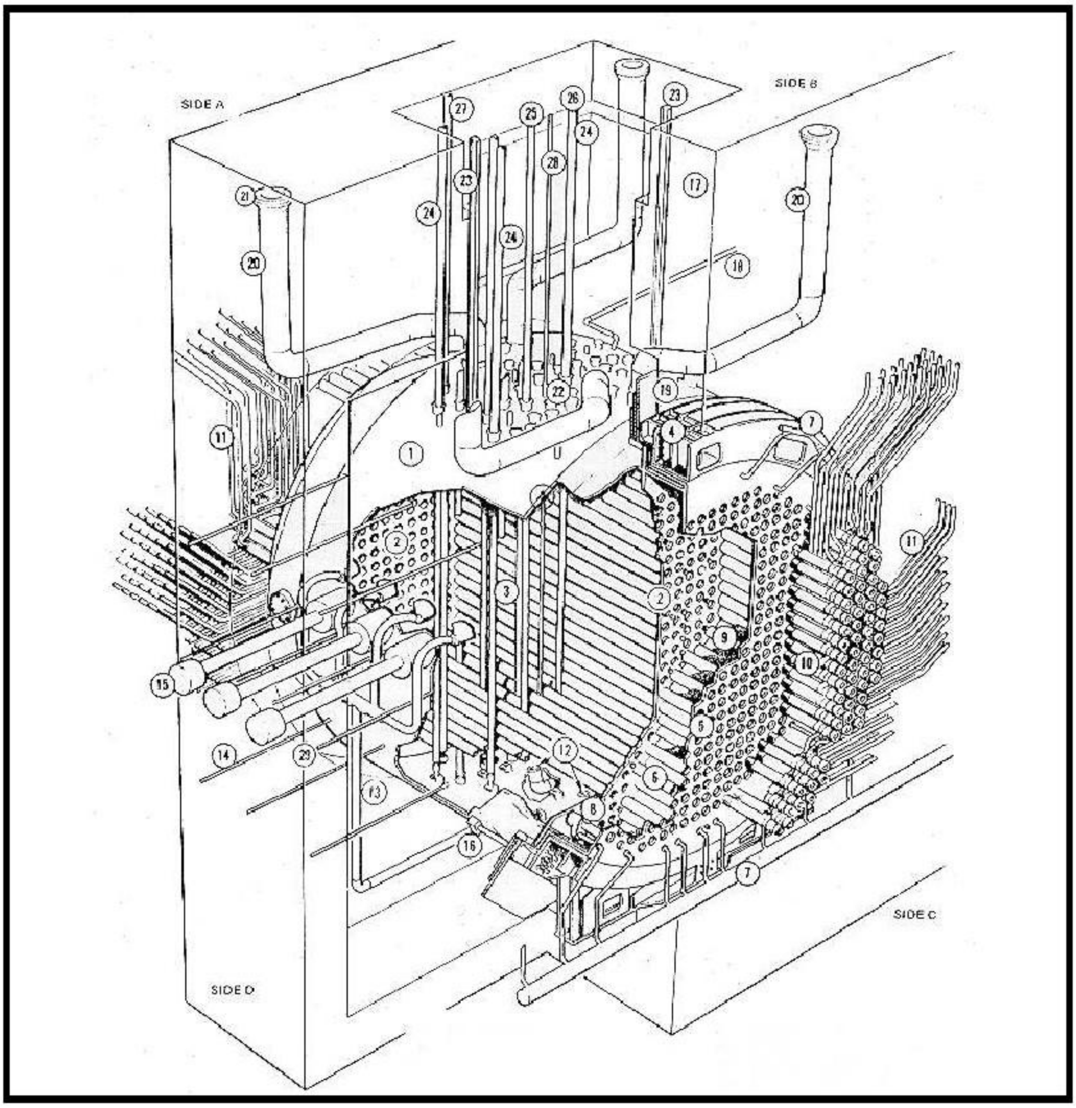


Figure V.2-4 - Embalse Nuclear Power Plant - Reactor Building

REACTOR ASSEMBLY



1. CALANDRIA
2. CALANDRIA-SIDE TUBESHEET
3. CALANDRIA TUBES
4. EMBEDMENT RING
5. FUELLING MACHINE - SIDE TUBESHEET
6. END SHIELD LATTICE TUBES
7. END SHIELD COOLING PIPES
8. INLET-OUTLET STRAINER
9. STEEL BALL SHIELDING
10. END FITTINGS
11. FEEDER PIPES
12. MODERATOR OUTLET
13. MODERATOR INLET
14. HORIZONTAL FLUX DETECTOR UNIT
15. ION CHAMBER
16. EARTHQUAKE RESTRAINT
17. CALANDRIA VAULT WALL
18. MODERATOR EXPANSION TO HEAD TANK
19. CURTAIN SHIELDING SLABS
20. PRESSURE RELIEF PIPES
21. RUPTURE DISC
22. REACTIVITY CONTROL UNIT NOZZLES
23. VIEWING PORT
24. SHUTOFF UNIT
25. ADJUSTER UNIT
26. CONTROLABSORBER UNIT
27. ZONE CONTROL UNIT
28. VERTICAL FLUX DETECTOR UNIT
29. LIQUID INJECTION SHUTDOWN NOZZLE

Figure V.2-5 - Embalse Nuclear Power Plant - Reactor Assembly

HEAT TRANSPORT SYSTEM NORMAL OPERATION FLOWSHEET

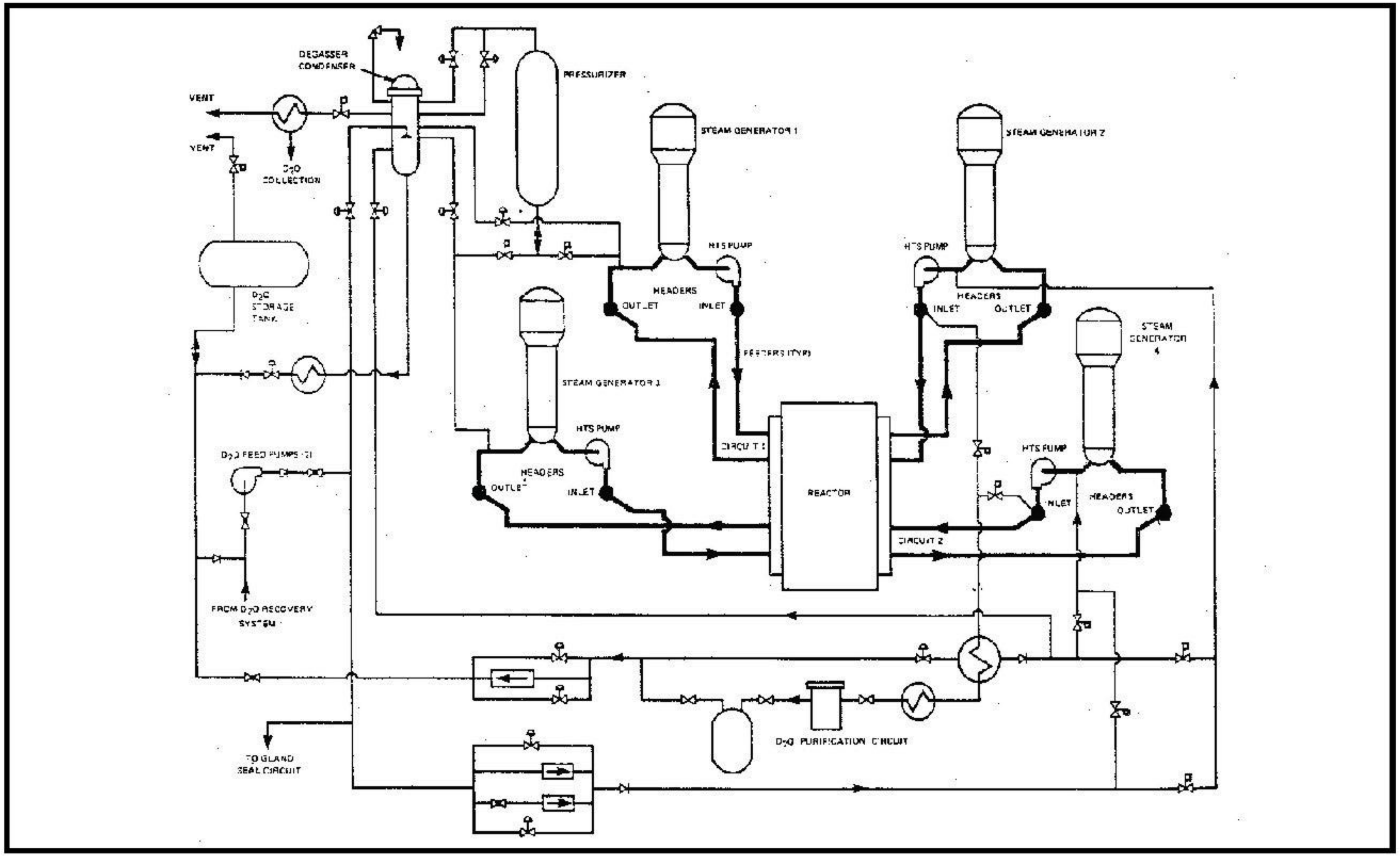


Figure V.2-6 - Embalse Nuclear Power Plant - Heat Transport System Normal Operation Flowsheet

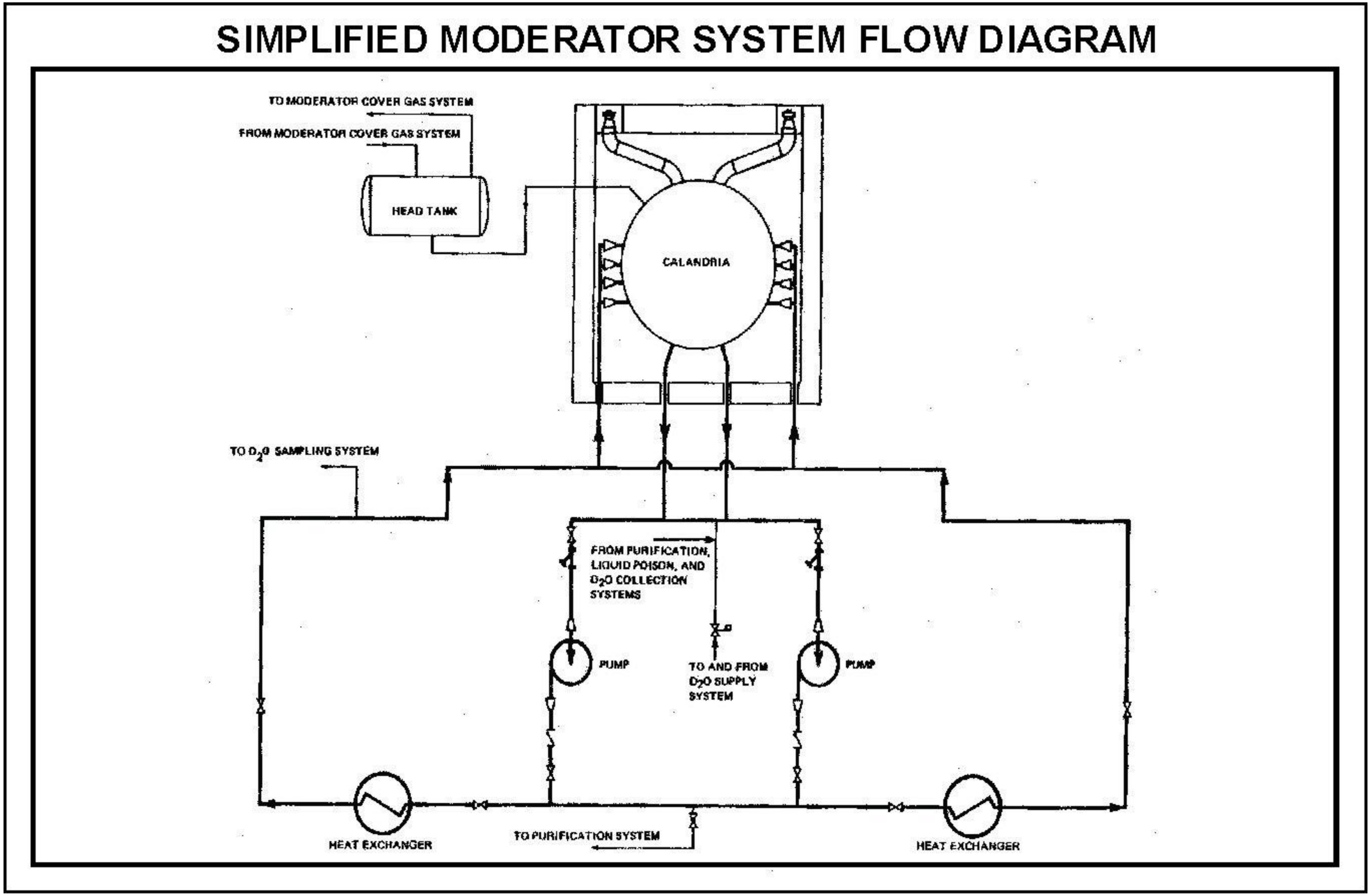


Figure V.2-7 - Embalse Nuclear Power Plant - Simplified Moderator System Flow Diagram

SIMPLIFIED SINGLE LINE ELECTRICAL DISTRIBUTION DIAGRAM

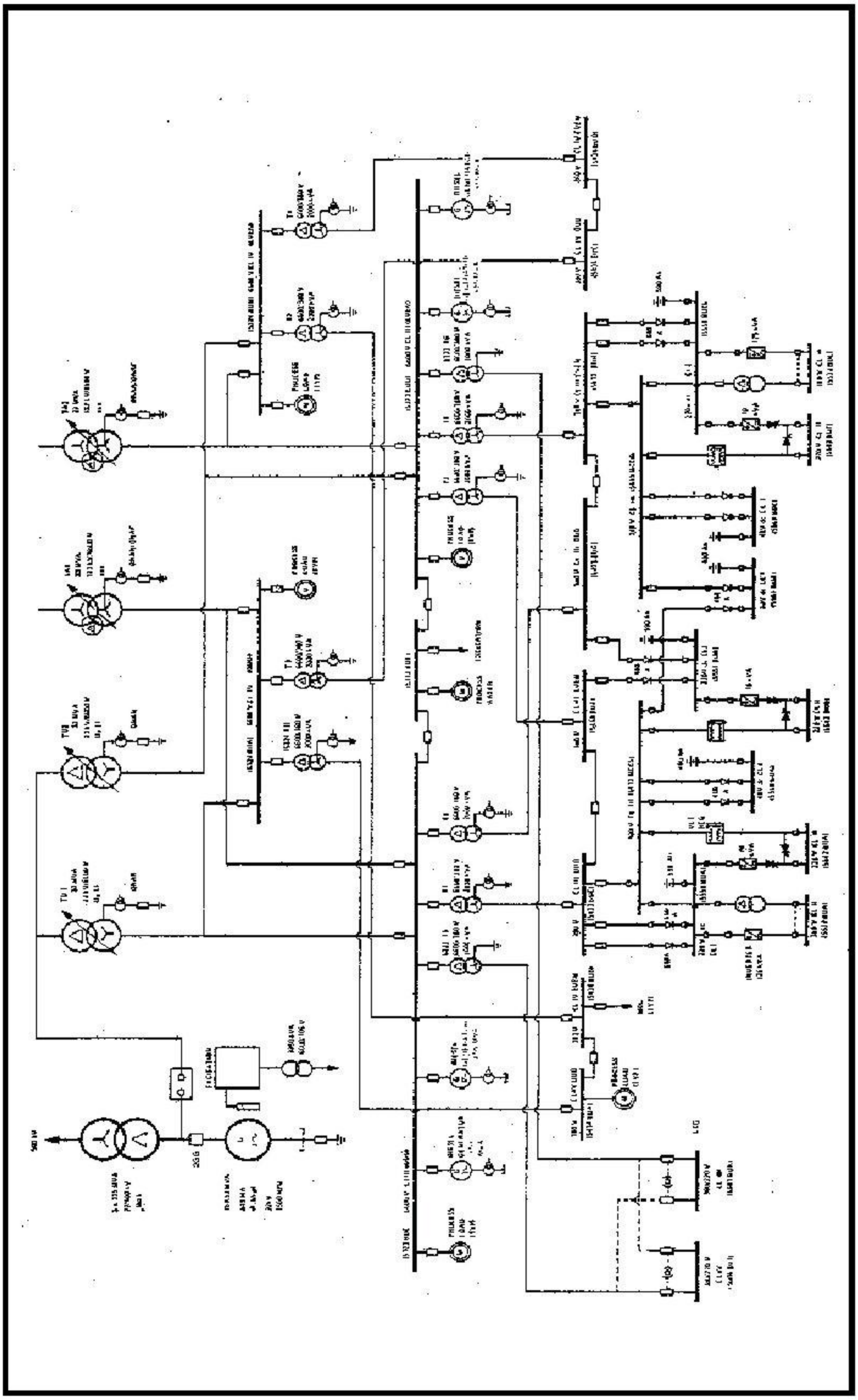


Figure V.2-8 - Embalse Nuclear Power Plant - Simplified Single Line Electrical Distribution Diagram

ANNEX VI

PRINCIPAL TECHNICAL CHARACTERISTICS OF ATUCHA II NUCLEAR POWER PLANT

VI.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 NASA, 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 VI-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 VI-2 shows schematically the main systems of CNA II.

VI.2. OVERALL PLANT LAYOUT

The overall layout ***(Figure VI-3)*** and arrangement of the CNA II on the site is governed by the following basic considerations:

- clear separation of the nuclear and conventional systems
- clear energy flow paths
- short piping and cable mans
- good transport conditions and access for construction, installation and operation

Buildings and structure arrangement of the CNA II are shown in Figure VI-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 roam, 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 occur.

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.

VI.3. CNA II MAIN SYSTEMS

VI.3.1 Reactor

The reactor ***(Figure VI-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 trigonal 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 pres-sure equalization openings of the moderator tank closure head. Therefore, the pressure differences in the core are comparatively small, which results in thin walls for the reactor pressure vessel internals. This allows a very high burn-up to be attained. Furthermore, the connection between the reactor coolant system and the moderator system permits the use of common auxiliary systems to maintain the necessary water quality. The number of auxiliary systems can therefore be reduced to a minimum.

For control of the reactivity, and thus of the power output of the reactor, various methods are applied. The reactor contains nine "blacks" {absorbers made of hafnium) and nine "grey" (steel) control elements arranged in 3 groups. The control elements are used to control the reactivity and the power distribution, to compensate the build-up of xenon poisoning following a reactor power reduction, to provide damping of azimuthal xenon oscillations, and to shut down the reactor. The reactivity value of all control elements is sufficient to shut the reactor.

In addition to the control, elements, reactivity control is provided by the boric acid dosing system. The injection or extraction of boric acid serves to compensate slow reactivity changes due to the burnup during the first period of operation and to maintain the reactor in a safe subcritical condition at zero power. Extraction of the boric acid is performed by anion exchangers.

Additionally, a boron injection system, as a second independent shutdown system is provided, which injects boric acid into the moderator.

The reactivity can, in addition to these reactivity control systems, also be controlled by varying the moderator temperature within a certain range, which is advantageous for some operating modes.

The reactor pressure vessel constitutes the pressure boundary of the reactor core and encloses the core components and the reactor pressure vessel internals. The RPV consists of the lower part, the closure head and the studs and nuts which connect both sections. The connection is made leak-tight by means of a welded lip seal.

The lower part of the RPV consists of the hemispherical bottom section, two shell courses and a shell flange which carries the coolant inlet and outlet nozzles and the support pads located between them. The reactor coolant inlet and outlet nozzles are arranged on one plane; there are no penetrations or pipe connections below this plane. The reactor core is housed below the plane of the inlet and outlet nozzles.

The closure head consists of a flange and a dome plate connected by a circumferential weld. The closure head dome carries the nozzles for coolant channels, moderator pipes, and control element drives and for in-core instrumentation. The nozzles are screwed into holes in the closure head dome and sealed by an overlay weld.

Most of the RPV internals form the structure of the reactor core. The moderator tank accommodates all core components, separates the moderator from the coolant and, in conjunction with the reactor pressure vessel, forms the annulus for, the in-flowing coolant. The bottom of the moderator tank serves as the lower fixing level for the coolant channels and the control element guide tubes. The moderator tank shell serves as thermal shielding.

The moderator tank closure head form the upper plenum for the reactor coolant leaving the coolant channel slots. The closure head and the moderator tank jointly form a unit which keeps the coolant channels and the control element guide tubes in position firmly and without displacement during all operating modes, as well as during postulated accidents. The moderator tank and its closure head are suspended from the flange of the reactor pressure vessel and are held tightly in position by the pressure vessel closure head.

The coolant channels consist of vertically arranged tubes which contain the fuel bundle columns, direct the reactor coolant flow and separate the reactor coolant from the surrounding moderator.

The reactor coolant flows inside the coolant channels in an upward direction. After passim through the fuel assembly, it leaves the coolant channel through slots and enters the upper plenum formed by the moderator tank closure head.

The coolant channel closure head, together with the coolant closure plug, forms the pressure-tight cap of the coolant channel. It can be opened by the refueling machine during reactor operation in order to exchange the fuel bundle column located inside the coolant channel.

The moderator piping serves for supply, distribution and extraction of the moderator inside the moderator tank. The moderator piping essentially encompasses four down-comers, the sparger ring on the moderator tank bottom, and the suction boxes with nozzles in the moderator tank closure head.

The moderator flows downwards through the down-comers to the sparger ring, where it is distributed at the moderator tank bottom. After rising and heat-up in the moderator tank, the moderator flows to the suction boxes and leaves the moderator tank through two nozzles.

The filler pieces are provided in the reactor pressure vessel in order to displace the heavy water and thus reduce the heavy water inventory rewired. The upper filler pieces are adapted to fit the reactor pressure vessel closure head. The lower filler pieces are divided into several interlocking rings and adapted to fit the bottom head of the pressure vessel.

VI.3.2 Reactor coolant system and moderator system

The reactor coolant system ***(Figure VI-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 surge lure, 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 beat 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.

VI.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 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 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.

VI.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.

VI.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.

VI.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.

VI.3.6 Electric power system

The CNA II has two physically independent grid connections. One of then is the 500 kV grid and the other is the 132 kV grid.

There is additionally, auxiliary power supply from the generator in case of grid disturbance after load rejection. Only in the case of a common outage of all three power supply possibilities, the emergency power system with the diesel generators will be required. Definite loads – mainly of the control and instrumentation system – are power supplied by rectifiers and converters or by means of batteries with direct current.

The generator feeds into the 500 kV network via one generator transformer and supplies the station auxiliary service requirements by means of two auxiliary transformers.

The four secondary windings of the auxiliary transformers or the two secondary windings of the offsite system transformer feed into four separate medium high voltage bus section (each two 6,6 kV and 13,2 kV), to which the large auxiliary loads and the transformers for the low voltage switchgears are connected.

If the station service power system fails, certain equipment (pumps, etc.) are needed to remove residual heat and to run the plant into safe conditions. This equipment must remain in operation or has to be put in operation and must therefore be supplied with emergency power. Here a distinction must be made between two groups with regard to safety related requirements: loads allowing a voltage interruption while the diesel run up and loads which must remain in operation without interruption or which must be put into operation immediately should the normal station service system fail.

The diesel emergency power system is like the other redundant safety equipment, divided into for separate trains. Under normal operating conditions, the auxiliary switchgears of the normal power system feed the emergency power system. To avoid loss of power in case the auxiliary power system fails, each of the four trains in the emergency power system is equipped with a quick-starting diesel set.

The CNA II electric system may be divided into two main subsystems: the offsite power system and the onsite power system.

The offsite power system is constituted by the 500 kV transmission line, which is linked to the substations Rosario Oeste (113 km), Colonia Elía (160 km) and Ezeioza (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.

VI.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 dawn and kept shut down over prolonged periods
- the decay heat can be reliably removed
- any release of radioactivity is within the limits laid down by the radiation protection regulations

In order to meet these requirements, safety measures against damage to the systems or components are provided. The safety measures can be classified under three safety levels according to the possible plant conditions:

- Components and systems necessary for normal operation (including startup, part load and full load operation, load changes, shutdown) are of such design as to preclude failure. The safety measures provided are:
    - conservative and careful design,
    - stringent quality assurance and control
    - regular examinations and inspections.
- According to general engineering experience, it must be considered that systems and components can fail during their service life despite adequately high quality. It is assumed that operational disturbances (e.g. reactor coolant pump failure, load rejection) can occur. In order to prevent the faults and operational disturbances and to mitigate their consequences the following safety measures are provided:
    - inherently safe operational characteristics
    - alarm annunciation
    - reactor protection limitation.

- Despite the safety measures of the first and second safety levels, theoretically assumed accidents are postulated. In order to counter these accidents and to mitigate their consequences, active safety systems are provided. The design of the safety systems is based on the assumption that parts of the safety systems (sub-systems) can fail simultaneously with the accident. As a consequence, the safety systems are of redundant design. This multiple-train design is reflected not only in the redundancy of the equipment, but also in the consistent physical segregation of the subsystems. This ensures that a sub-system failure (random failure), postulated in addition to the accident, remains restricted to one subsystem and does not affect the others.

According to the design principles of the safety systems all engineered safety features necessary to control accidents are built as four identical sub-systems. Two of these are sufficient for the control of the accident. Thus, functional availability is assured even when one subsystem is being inspected or repaired and a single failure occurs simultaneously in another subsystem. Each subsystem also comprises the associated power supply and the necessary auxiliary equipment.

The basic safety systems provided are:

- Fast reactor shutdown system
- Emergency core cooling system
- Containment system
- Emergency electric power system

In order to protect the environment against the release of radioactivity, the following radioactivity barriers are provided as passive safety measures:

- the fuel matrix of the uranium dioxide pellets
- the seal-welded cladding tubes enclosing the fuel
- the closed and seal-welded reactor coolant system and moderator system
- the full pressure gas tight steel containment structure
- the concrete secondary shield

The components of the radioactivity barriers act according to their mechanical properties, without auxiliary energy. In case of damage to one of these barriers, the next one will act and thus retain the radioactivity.

The accidents considered in the plant design are the plant internal and external accidents. The internal accidents are, above all, LOCA, with the whole spectrum of pipe ruptures including the break of the largest connection pipe to the reactor coolant loops or to the moderator system. The external accidents considered are aeroplane crash, explosion pressure wave, floods, tornadoes, etc.

In order to meet the safety requirements even during the considered internal and external accidents, the following design principles were established:

- multiplicity of safety features
- redundancy of the safety systems and of their auxiliary system
- diversity of important parts of the reactor protection system
- physical separation and/or protection by concrete walls of the redundant sub-systems
- protection of the safety systems against external accidents
- periodic testing of the safety systems.

The task of the safety systems is to prevent any damage to the radioactivity barriers during operational malfunctions and during accidents in order to fulfill 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 VI-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 VI.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.

VI.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	312º C
Coolant temperature rise though the core	34º C
Moderator temperature normal/maximum	170º/220º 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	271º C
Total steam flow	3445 Ton/h

Steel Containment Data	
Diameter	56 m
Wall thickness	30 mm
Design pressure	5.3 bar
Design leak rate	0.25 vol %/day

SITE LOCATION

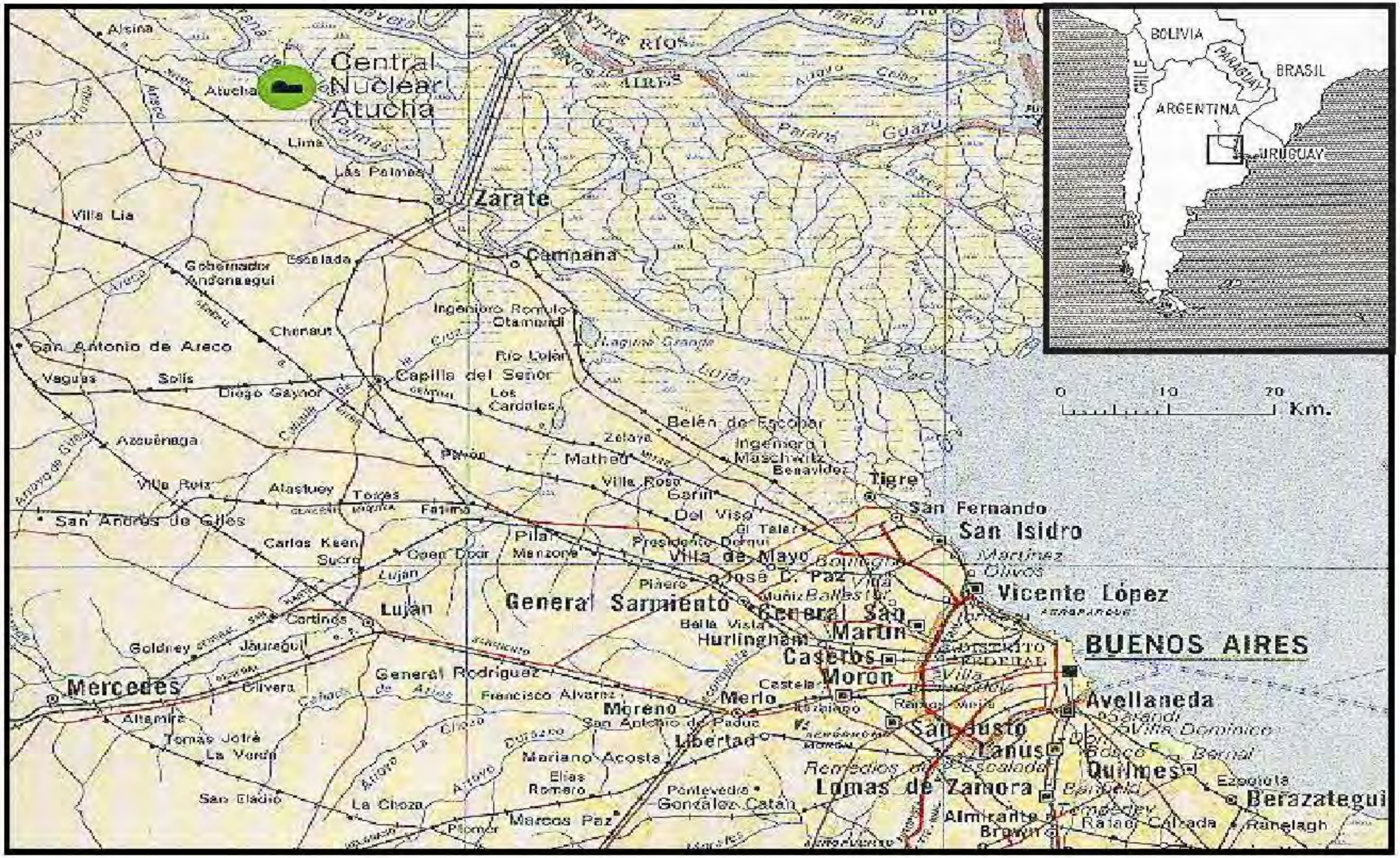


Figure VI-1 - Atucha II Nuclear Power Plant - Geographic Location

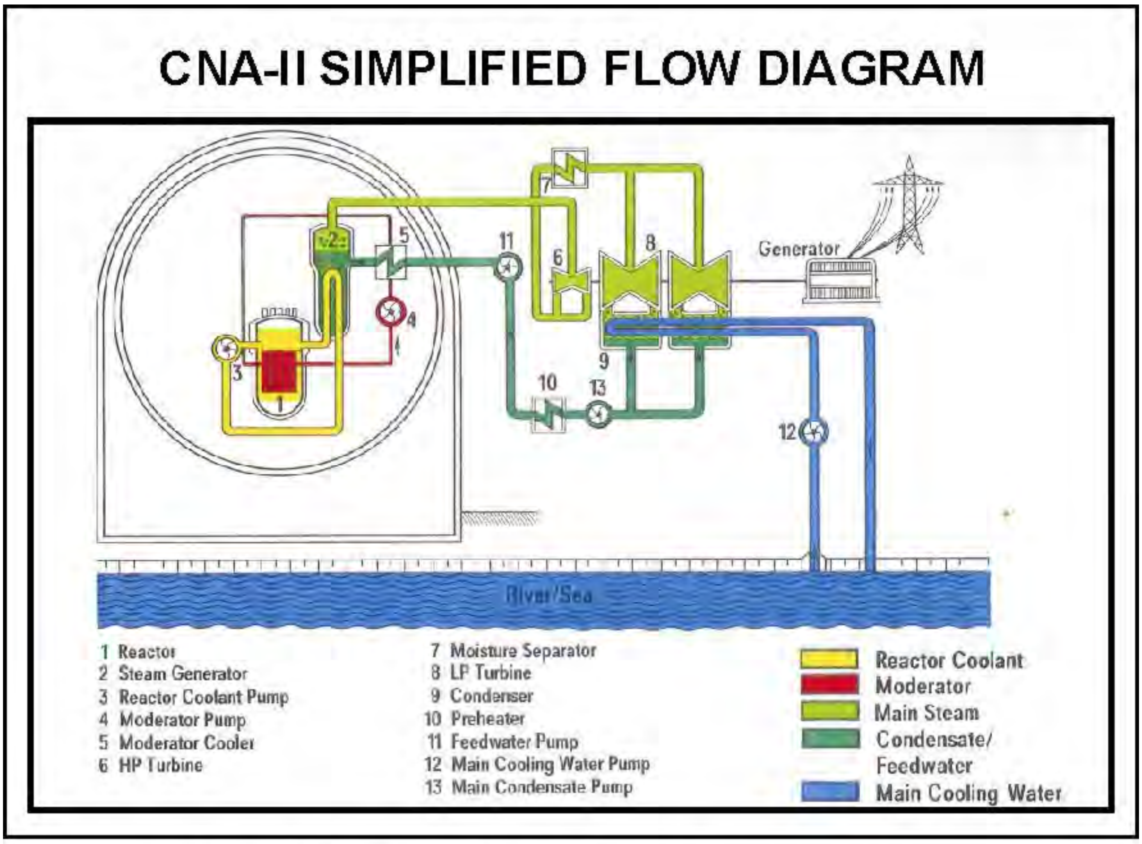


Figure VI-2 - Atucha II Nuclear Power Plant - Main Systems

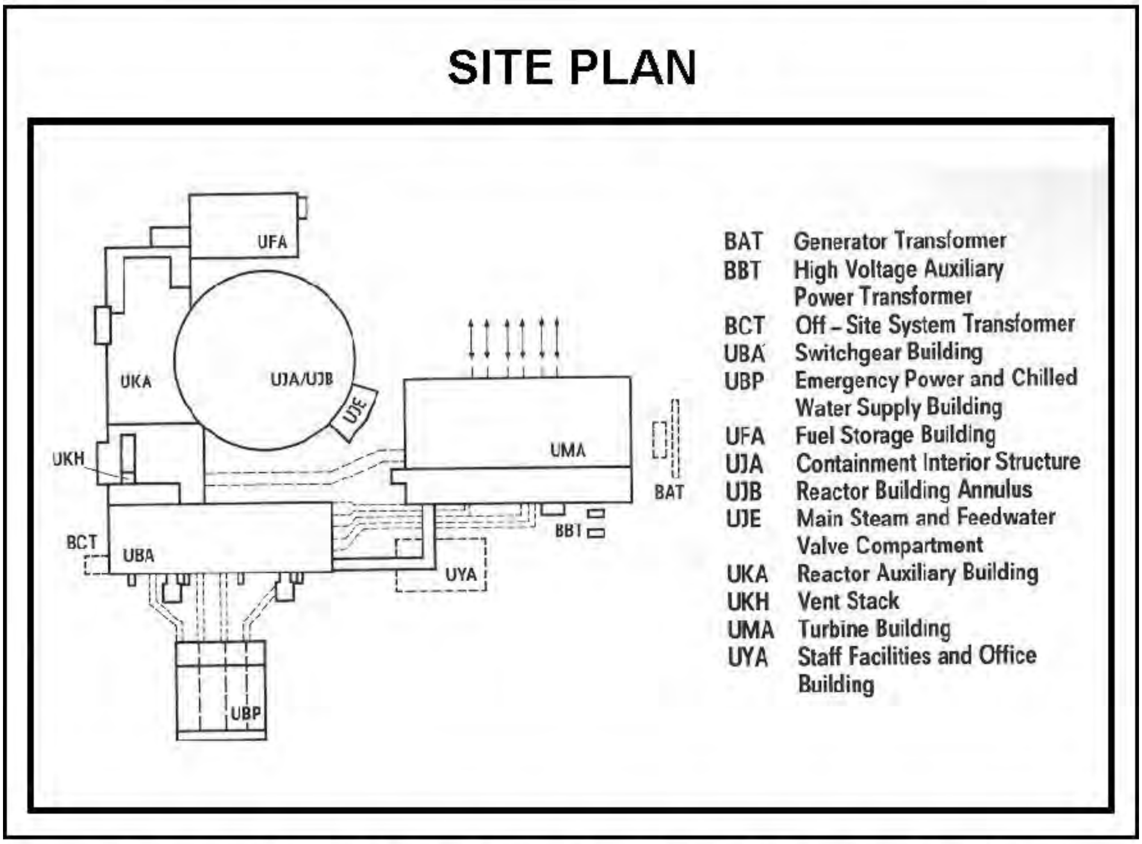


Figure VI-3 - Atucha II Nuclear Power Plant - Main Buildings and Structures

REACTOR PRESSURE VESSEL - INTERNALS

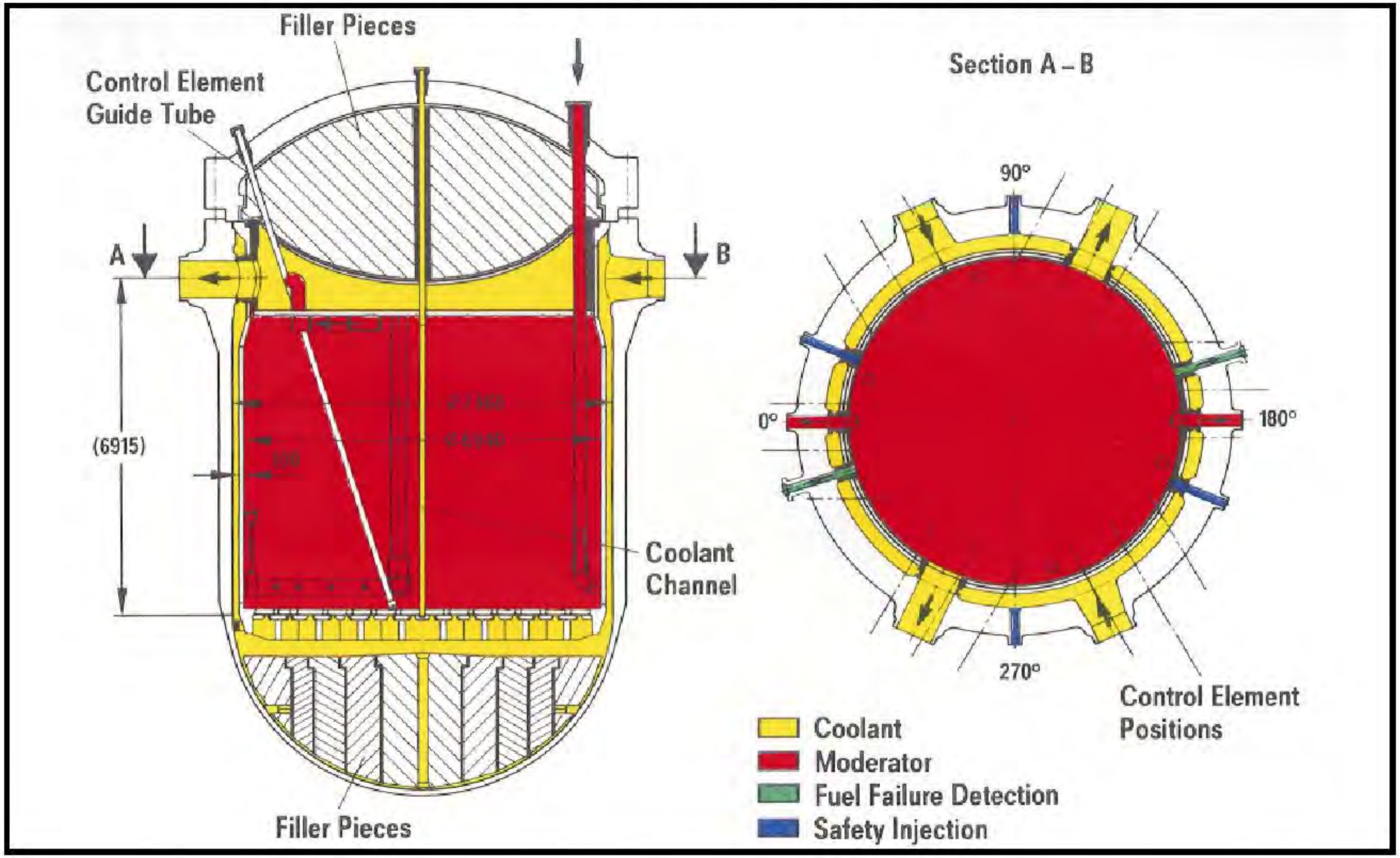


Figure VI-4 - Atucha II Nuclear Power Plant - Reactor Pressure Vessel with Internals

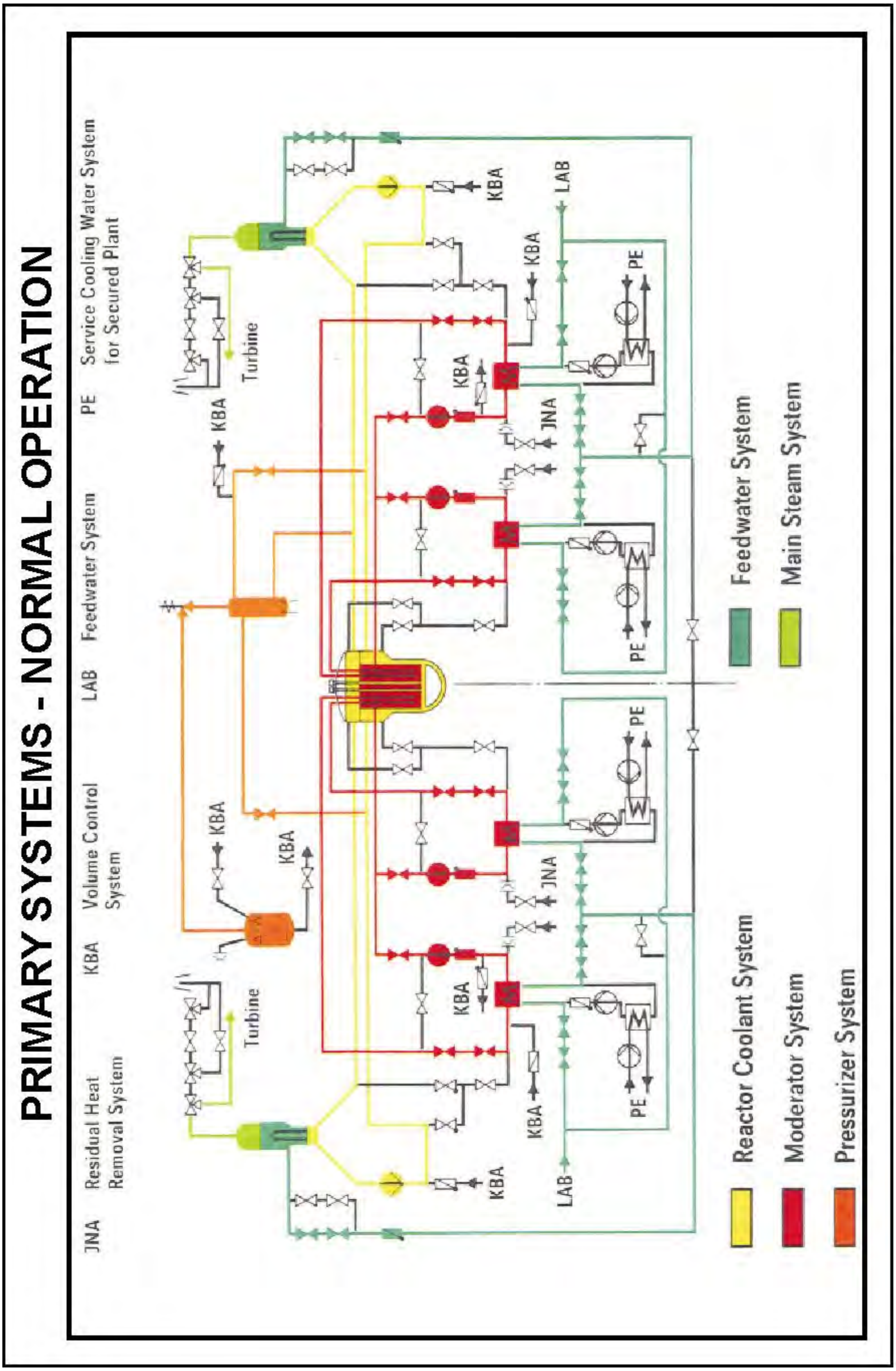


Figure VI-5 - Atucha II Nuclear Power Plant - Primary Systems - Normal Operation

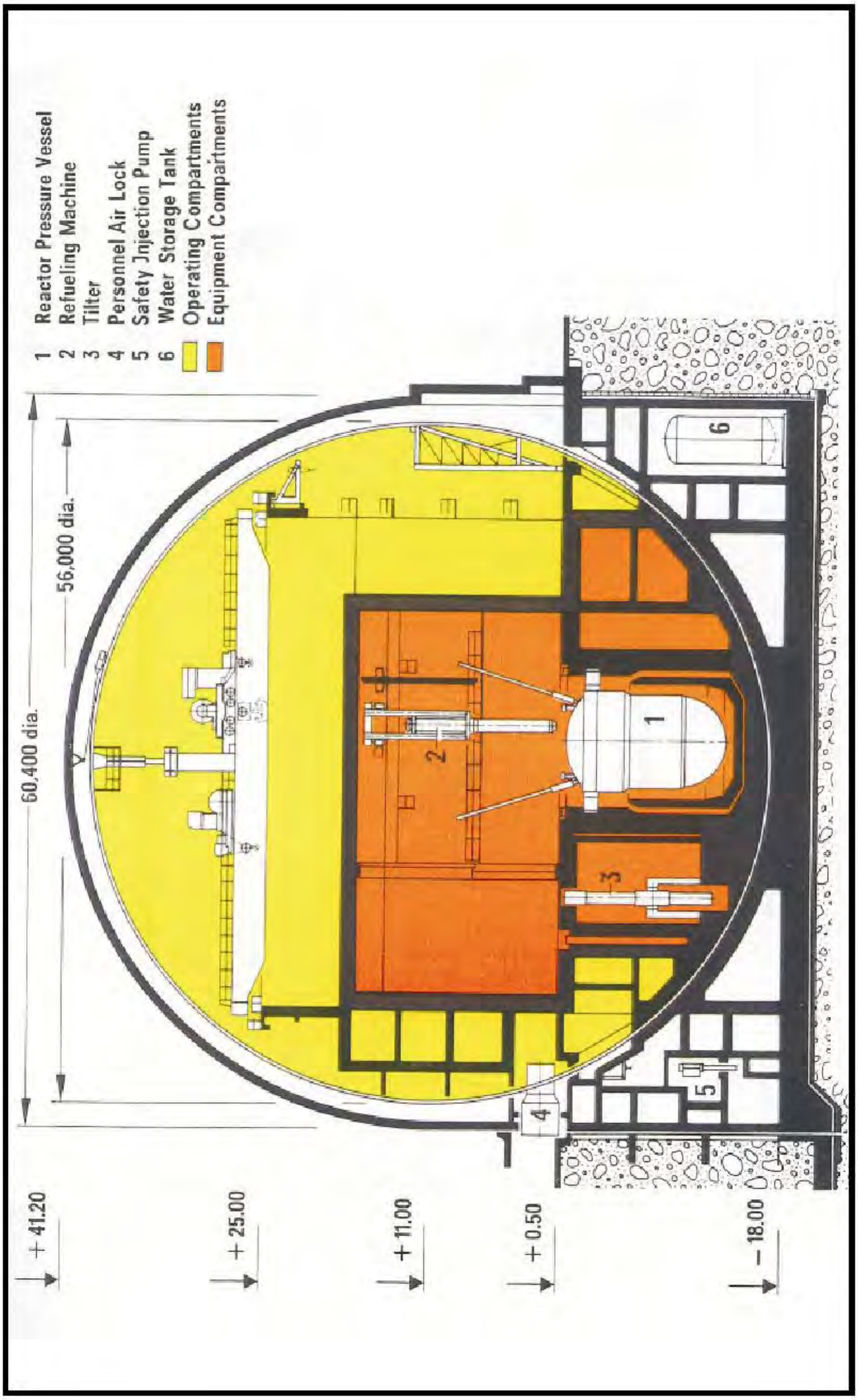


Figure VI-6 - Atucha II Nuclear Power Plant - Reactor Building

ANNEX VII
PRINCIPAL TECHNICAL CHARACTERISTICS OF CAREM REACTOR PROTOTYPE

VII.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.

VII.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 VII-1).***

The flow rate in the reactor primary systems is achieved by natural circulation. Figure VII-1 shows a diagram of the natural circulation of the coolant in the primary system. Water enters the core from the lower plenum. After been 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.

VII.3. REACTOR CORE AND FUEL DESIGN

The core has Fuel Assemblies (FA) of hexagonal cross section. Each fuel assembly contains 108 fuel rods of 9mm outer diameter, 18 guide thimbles and 1 instrumentation thimble ***(Figure VII-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 330 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 to produce a sudden interruption of the nuclear chain reaction when required (fast shutdown system).

VII.4. STEAM GENERATORS

Twelve identical ‘Mini-helical’ vertical steam generators, of the “once-through” type are placed equally distant from each other along the inner surface of the RPV ***(Figure VII-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.

An external shell surrounding the outer coil layer and adequate seal form the flow separation system. It guarantees that the entire stream of 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.

VII.5. REACTOR AUXILIARY SYSTEMS

Figure VII-4 shows a diagram of the main reactor auxiliary systems c*hemical / 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 solid way at 45°C until full power operation.

VII.5.1 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 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.

VII.5.2 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, each branch comprising a pump, plate heat exchanger for cooling, and shell and tube heat exchanger for heating.

VII.5.3 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 and pumps.

VII.5.4 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, sharing the filter and the ion exchange bed for water purification.

VII.5.5 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.

VII.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 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.

VII.7. TURBINE GENERATOR PLANT SYSTEMS

The CAREM has a standard steam cycle of simple design. The steam generators are built as a drum-less "once-through" boiler without accumulators between them and the consumer equipment. In accordance with the behaviour of once-through boilers, steam is superheated in every plant condition. No super-heater is needed. Likewise, no blow-down is needed in the steam generators, reducing the waste generation. 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. The condensate is pumped and sent to the full stream polishing system in order to maintain ultra-pure water conditions. High purity water leaving the polishing system is sent to the low-pressure pre-heater using turbine extraction as a heating media. The warm water is delivered to the water accumulator in order to perform degassing operations with additional heating using extraction steam. Water is then pumped to the high-pressure pre-heaters (two in tandem using extraction steam) and sent to the steam generators as a feed-water closing the circuit.

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.

VII.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 hierarchic 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.

VII.9. REACTOR PROTECTION SYSTEM

The design of the reactor protection system was performed according with the most advanced technology for nuclear power plants design, the "defence in depth" principle and the early failure detection, with the object of avoid the occurrence of accidents beyond the design base.

The reactor protection system has two independent subsystems. The first subsystem, responsible of the generation of the first shutdown system trip signal, consists in a combination of hard logic and digital processing modules. The second subsystem, responsible of the generation of the second shutdown system trip signal, is based in a hard logic technology in order to fulfil with the 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,
- Failure tolerance,
- Possible in operation testing,
- Safe failure.

The interaction between the protection system and the control system is performed through electrical isolation. The interfaces are design 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.

VII.10. ELECTRICAL SYSTEMS

The electrical loads are divided in three classes:

- Class I: DC, 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 and III correspond to the Safety-related system. Class IV includes all the conventional systems. The electrical power supply corresponding to class I and III systems are distributed by two systems of independent buses. This redounded 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 redounded, physically separated and they can supply each of the power distribution systems of classes I and III.

Class I is sized to supply power to selected safety-related loads for at least 48 hours before needing a connection to classes III, IV or other external power source.

VII.11. PLANT LAYOUT

CAREM nuclear island is placed inside a pressure suppression containment system, which contains the energy and prevents fission product release in the event of accidents **(Figure VII-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

VII.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 act 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 5 levels. In the upper part of this box are the Fuel Elements Pool and the Auxiliary Pool, and in the lower part are the Liquid Effluent and Spent Resin Pools. In CAREM-25 these five levels are:

VII.11.1.1 Level + 15.20: The Reactor Hall where tasks related with the refuelling will be performed.

VII.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.

VII.11.1.3 Level +5.20: The Gaseous Waste treatment system, valves rooms for the filters and resins beds, pumps and heat exchangers for the Components Cooling System.

VII.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, and the HVAC injection equipment

VII.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.

VII.11.1.6 Containment: The containment is divided in two main compartments: a drywell and a wetwell.

The upper drywell 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, below the upper level and houses the pipelines connected to the SG's. The Wetwell (below the Peripheral Drywell, and surrounding the Central Drywell) is partially filled with water, conforming the Pressure Suppression Pool. The tube vents are immersed deep in the pool and they connect to the floor of the Peripheral Drywell.

VII.11.2 Turbine module

It houses the turbo-generator group, auxiliary services like 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.

VII.11.3 Control module

The Control Complex is placed in this area of the building. It is formed by the Cables 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.

VII.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 two days.

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, is contained inside a single pressure vessel. This considerably reduces the number of pressure vessels and simplifies the layout. Due to the absence of large diameter piping associated to the primary system, no large LOCA has to be handled by the safety systems. The elimination of large LOCA 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 like 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.

VII.13. SAFETY SYSTEMS AND FEATURES

The safety systems are duplicated to fulfil the redundancy criteria ***(Figure VII-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. Each neutron absorbing element is a cluster composed of a maximum of 18 individual rods which are together in a single unit. Each unit fits well into guide tubes of each fuel assembly.

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. Six out of twenty-five CRD (simplified operating diagrams are shown in ***Figure VII-7***) are the Fast Shutdown System. During normal operation they are kept in the upper position, where the piston partially closes the outlet orifice and reduces the water flow to a leakage into the RPV dome. The CRD of the Adjust and Control System is a hinged device, controlled in steps fixed in position by pulses over a base flow, designed to guarantee that each pulse will produce only one step.

Both types of devices perform the SCRAM function by the same principle: “rod drops by gravity when flow is interrupted”, so malfunction of any powered part of the hydraulic circuit (i.e. valve or pump failures) will cause the immediate shutdown of the reactor. CRD of the Fast Shutdown System is designed using a large gap between piston and cylinder in order to obtain a minimum dropping time thus taking few seconds to insert absorbing rods completely inside the core. For the Adjust and Control System CRD manufacturing and assembling allowances are stricter and clearances are narrower, but there is no stringent requirement on dropping time.

The Second Shutdown System (SSS) is a gravity-driven injection device of borated water at high pressure. It actuates automatically when the Reactor Protection System detects the failure of the First Shutdown System or in case of LOCA. 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. The tanks are pressurised, thus when during a LOCA the pressure in the reactor vessel reaches a relative low pressure, rupture disks break and the flooding of the RPV starts, preventing core un-covery for a long period. The R*esidual Heat Removal System* is also triggered to help to depressurise the primary system, in case the breakage area is small.

Three *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.

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 is of 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 VII.I.

Table VII.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
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 steams generators, the Residual Heat Removal System is demanded cooling the primary system reducing reactor pressure to values lower than the ones of hot shutdown. In case of hypothetical failure of FSS, the reactor power reduces due to the negative reactivity coefficients without compromising the fuel elements. The SSS will guarantee medium and long-term reactor shutdown.

Total loss of flow: due to the absence of primary pumps, this initiating event is avoided.

Loss of coolant accident: RPV penetration maximum diameter is limited by design, therefore no large LOCA is possible and there is no need of a high-pressure injection system. In case of LOCA the FSS, SSS, RHRS are demanded and when pressure decreases the *Emergency Injection System* discharge water to keep the core covered for several days. As no credit is given by design to active systems, the secondary system is not considered to cool and depressurise the primary system in safety evaluations, of course if it is available and in case of need it could be used as part of Accident Management Strategy. Moreover, by design no credit is given to a broken pipe as an injection line (steam coming into the RPV from the containment in case of high depressurisation of the primary system due to the use of the steam generators). The reactor inherent response to LOCA was also analyzed, considering FSS success and failure of all the Safety Systems related with core cooling. Due to the large water inventory over the core and the small penetration diameters through the RPV, the core uncovers after several hours.

Steam generator tube rupture: this accident is mitigated by isolating the group of steam generators affected, closing both the steam and feedwater lines. The secondary side of the steam generators reaches thermal equilibrium with the primary circuit, equalising pressure with this system. Eventually the reactor could continue operating at 50% of power.

Steam line break accident: The sudden depressurisation of the secondary side of the steam generators increase heat removal from the primary system with the consequent core overpower. Reactor shutdown (FSS and SSS) and Residual Heat Removal System are demanded and the reactor reaches a safe condition. In case of hypothetical failure of both shutdown systems, reactor overpower does no compromise safety critical values (DNB and CPR) because primary total heat removal by the steam generators is intrinsically limited by the reduced tube side water inventory.

Blackout: 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 - two days), 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.

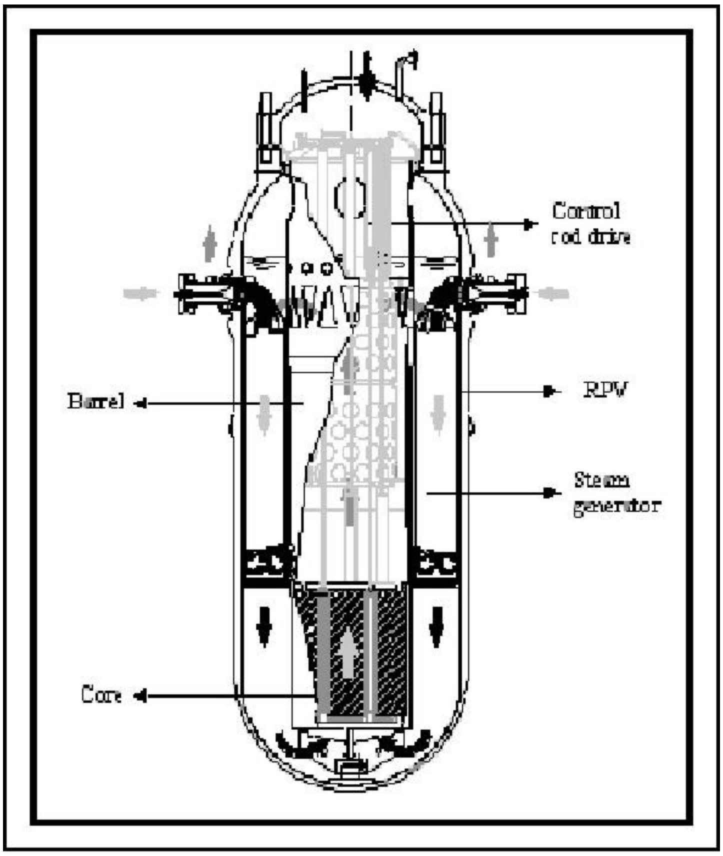


Figure VII-1 - Reactor Pressure Vessel

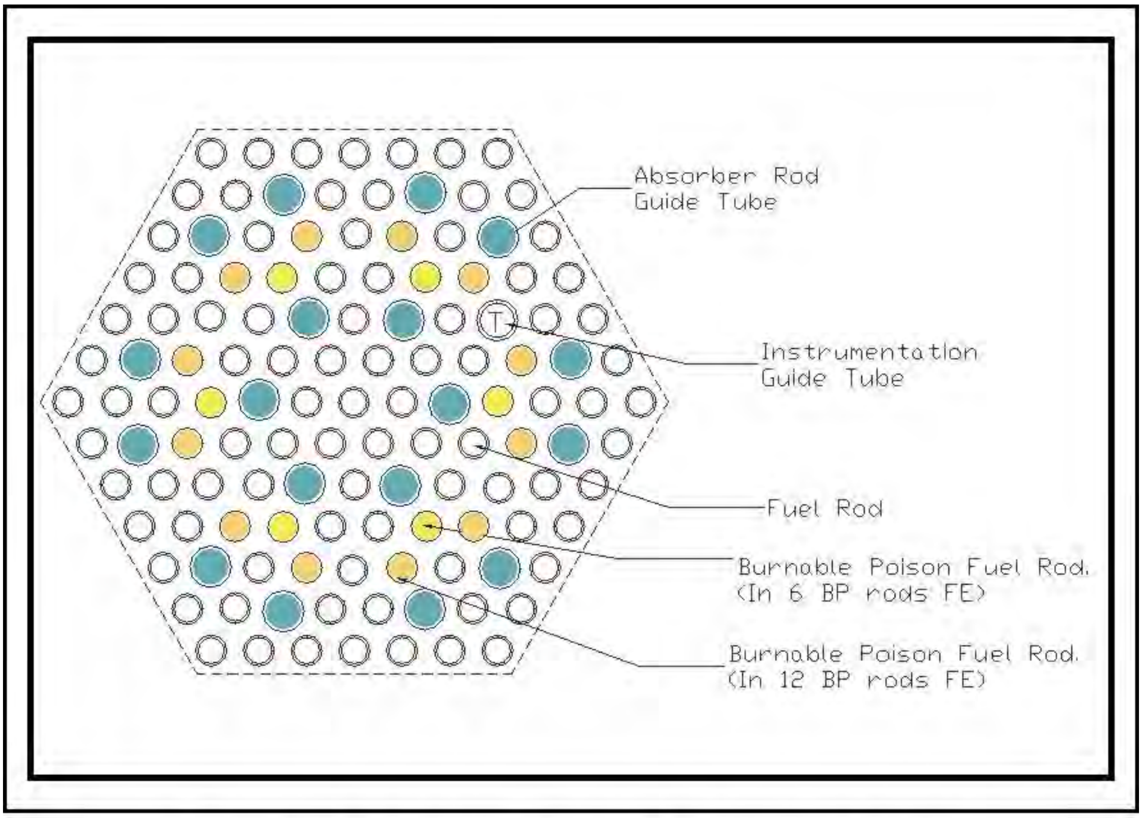


Figure VII-2 - Fuel Assembly Diagram. Fuel rods, guide thimbles and instrumentation thimble distribution

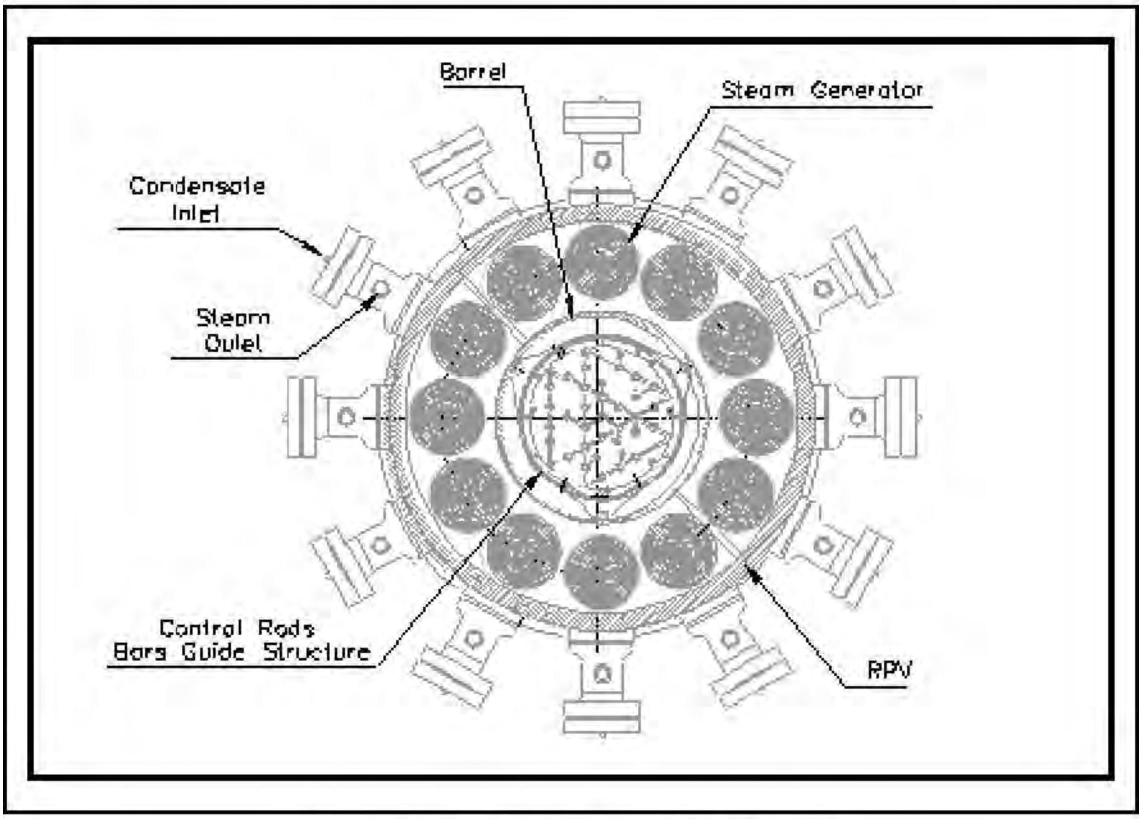


Figure VII-3 - Steam Generation Layout

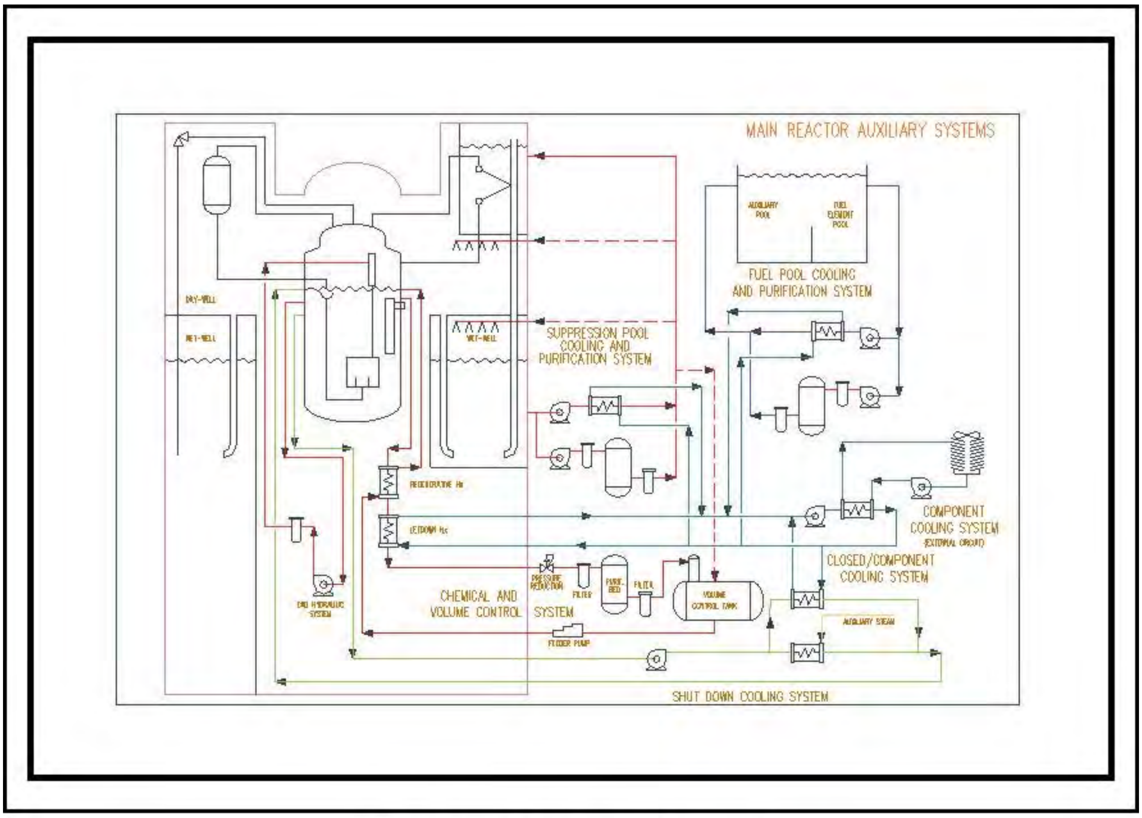
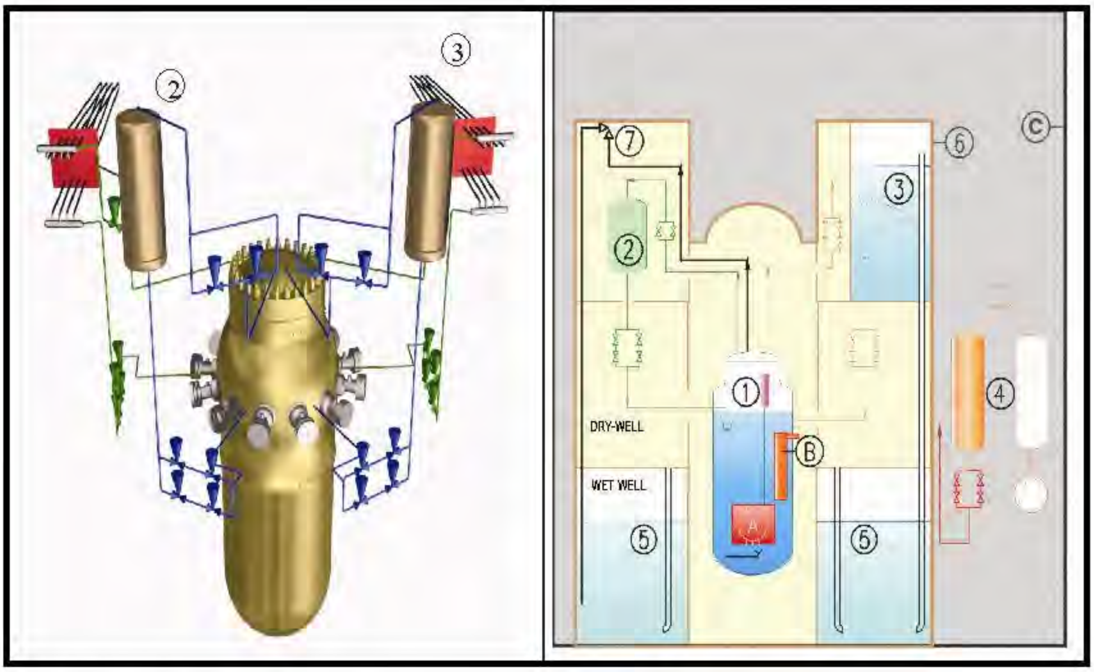


Figure VII-4 - Auxiliary Systems

Figure VII-5 - Plant Layout



1- First Shutdown System
2- Second Shutdown System
3- Residual Heat Removal System
4- Emergency Injection System
5- Pressure Suppression Pool
6- Containment
7- Safety Valves
A- Core
B- Steam Generators
C- Reactor Building

Figure VII-6 - Containment and Safety Systems

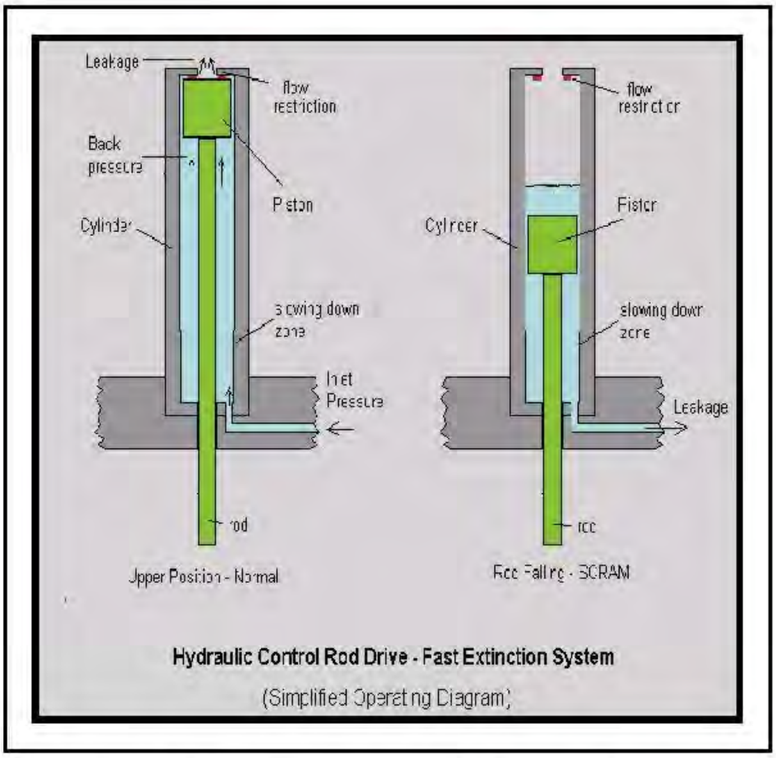


Figure VII-7 - Simplified Operating Diagram of a Hydraulic Control Rod Drive (Fast Shutdown System)

ANNEX VIII

EXAMPLES OF LESSON LEARNED AND CORRECTIVE ACTIONS RESULTING FROM NATIONAL AND INTERNATIONAL OPERATING EXPERIENCE AND EVENTS

The Operative Experience comming from National and International events are given in this Annex.

Besides, the most significant operational events in CNA I and CNE in the period 2004 – 2007,and how the licensees and the ARN have consequently and accordingly acted, are shown.

VIII.1. EXAMPLES OF LESSON LEARNED AND CORRECTIVE ACTIONS RESULTING FROM INTERNATIONAL EVENTS AND OPERATING EXPERIENCE

EVENT - OPERATING EXPERIENCE	EXAMPLES OF LESSONS LEARNED - CORRECTIVE ACTIONS
Darlington 3 CANDU NPP Neutron Dose Assessment. Detection of neutrons in an area of the R/B that is accessible during normal operation (near airlocks).	CNE specific measurements were implemented for works done near airlocks.
Point Lepreau. CANDU NPP Steam leak on the condenser steam discharge valve dump header	-Used as “lessons learned” belong the Flow Assisted Corrosion Program. -Inspection tasks were improved during 2005 Planned Outage
Main Output Transformer (MOT) Deluge Valve Failed Full Flow Test (COG, OER 1653 DA 04)	-OPEX Group Recommended revision / modification, when it would be necessary, valves test / inspection procedures. -Used as “lessons learned”. -Deluge valve inspection test procedure revised at CNE -Deluge valve test frequency changed from annual to biannual at CNA II NPP
Undetermined Unavailability of ECCS Strainers in the reactor building basement (WANO MER ATL 06-0024)	-The procedure for reactor building basement inspection already exist, however, it would be include in “restart program after outages”. -Use of the procedure every time that materials or equipment have been used in different tasks in reactor building basement
Corrosion of steel containment and containment liner (NRC IN 2004-09)	-Visual Inspection of Containment to identify possible corrective actions (surveillance or repairing). -Complementary inspections to repetitive tests were added.
Potential explosion hazard near hydrogen addition panels COG OER: 1446 PA 04	-Verification of potential explosion sources around hydrogen bottles bank at CNE -Lesson to be included in design, construction and commissioning of CNA II
Pipe break in Essential Service Water System WANO EAR PAR 05-039 (INES Level 2)	-Improvement of the service water inspections including in it the buried pipes (CNE)

EVENT - OPERATING EXPERIENCE	EXAMPLES OF LESSONS LEARNED - CORRECTIVE ACTIONS
Adjuster rods stuck in a partially withdrawn position – Reactivity Management COG OER: 1830 CV 05	-Revision of the Adjuster Rods Motor Variable Frequency Inverter calibration procedure including commissioning tests result (CNE)
Unexpected wear at first stage reheater steam piping COG OER: 1982 DA 05	-Similar events were reviewed for Flow Assisted Corrosion inspection program (CNE) -Used as “lessons learned”
Loss of normal water level in the radioactive liquid waste tank.	-Operation management weakness: equipment/system not used as designed -Lack of questioning attitude -Use the lessons learned to the management staff training
Loviisa 1. PWR. Ageing of cables en SGs rooms	Preventive manteinance using thermograthy.

VIII.2. MOST SIGNIFICANT OPERATIONAL EVENTS IN CNA I AND CNE. LESSON LEARNED AND CORRECTIVE ACTIONS RESULTING FROM NATIONAL EVENTS. OPERATING EXPERIENCE.

VIII.2.1 Loss of off-site power (CNA I, Feb 24 2004) (Reported as WANO EAR PAR 05-027)

On 02/24/04, Atucha 1 NPP suffered a completely loss of external electrical power supply (LOSP) due to an electric short-circuit in the transmission line.

The 220 kV breakers opened at Villa Lía substation and as a result there were not possibility to feed out the station power. A failure has occurred after a 220 kV electric breaker maintenance at this substation. The electrical grid configuration at this time was unable to assure the safety electrical supply to CNA 1.

There was a reactor automatic shutdown (SCRAM) due to faulty step power relay.

During the event, the station diesel generators provided emergency electrical supply.

The off-site power was recovered after 13 minutes.

All the station safety components operated as designed.

Lessons learned / Corrective actions:

- If the 220 kV transmission lines are lost, open immediately the 220/132 kV self-transformer primary side breakers to allow generator fed his own loads. Power feed from 132 kV line is not recommended.
- Agreement with grid operator to modify the 220 kV bus bar protections (implement PFI concept: protection against breaker failure)
- Change step power relay calibration set point

VIII.2.2 Fuel Elements failure CNA I, 2004 (Reported as WANO EAR PAR 05-022)

CNA I´s original fuel elements consisted of bundles containing natural uranium. In 1995 a slightly enriched uranium (SEU) program commenced. The replacement of the complete core with these new FE was completed in 2000. These 0.85% enriched uranium FEs are supplied by the same manufacturer.

SEU fuel elements failures were at an acceptable level from the beginning of the program until 2004. In October 2004 a radiochemical primary system parameter increase was observed. A FE failure program was implemented in order to detect and take out failed FE from the core. The fuel elements that failed were manufactured with uranium of the same shipment and came from a mixture of natural uranium with pellets of 2.5% enriched uranium.

The Regulatory Body participated in the meetings that the Licensee had with the manufacturer, experts from the CNEA and CEN/SCK (Belgium) and requested all the information about the actions taken in order to find the causes of fuel failure and those concerning fuel management at the plant.

The analysis did not find an unambiguous cause. The Regulatory Body then requested an independent technical audit on the engineering and manufacturing process. NASA made a contract with the institute SCK-CEN from Belgium and the mission came in 2006.

<u>Lessons learned / Corrective actions:</u>

The manufacturer introduced some corrections, changes and improvements in the fabrication process that normalized the situation returning the failure rate to the historical values.

VIII.2.3 Unforeseen personnel exposure (CNA I September 2005)

During CNA I's fuelling machine maintenance tasks there was a heavy water spill. As a consequence some workers received a contamination due to tritium intake. One worker had an effective dose of *41,85 mSv.* That person already had an accumulated dose of 12.63 mSv in one year and 92.13 mSv in five years so as a result of this event his dose exceeds the annual and five year doses according to the present standards. According with what it is established in the Operating License, the Primary responsible communicated the incident to ARN.

The Regulatory Body, according to internal procedures, designed a group to make the root cause analysis of the event, which found that there were non compliances with procedures, failures in controls made after maintenance works, failures in calibrations and poor communication among operator shifts and among the shift personnel.

The plant also constituted an internal committee that analysed the event and suggested a set of improvements to be applied in the plant.

<u>Lessons learned / Corrective actions:</u>

There were several changes made in the organization, new procedures and retraining of personnel and existing documents were modified, as a result of ARN requirements and the internal plant committee's proposals of improvement.

The Licensee also decided some changes in the maximum levels of the NPP's organization.

VIII.2.4 Fuel element drops in Fuel Storage Pool (CNA I, Dec 27 2005) (Reported as WANO EAR PAR 06-025)

While trying to place a semi-burnt fuel element in an auxiliary hanger inside the pool of the Second Spent Fuel Building, it fell inside the fuel storage pool. This was an infrequent task in which the fuel element should be relocated in the reactor. As a result of its impact to the bottom of the pool, the fuel element was damaged and liberated Noble Gases, especially Xenon 133 isotope.

The fall was produced by an inappropriate manoeuvring of the rigging pole from the hanger position, after placing the fuel element and consequently it was dragged.

<u>Lessons learned / Corrective actions:</u>

- Disseminate this event to all Operation personnel, highlighting the importance of pre-job meetings.
- Emphasize the fact that periodical checks are necessary during the fuel element manoeuvres and in the rigging element itself

VIII.2.5 Primary Heat Transport System - Liquid Relief Valve Opening (CNE Jul 20 2004)

CNE was operating normally at full power when it was observed that the liquid relief valve 3332 PV13's opening caused automatic reactor power reduction to reach the cold shutdown condition. The safety systems were not activated. CNE was 48 hours out of service.

During the plant outage, corrective actions to restore normal liquid relief valve condition were performed by replacing the failed liquid relief valve's diaphragm and then the plant was re-started again. It is a recurrent event in CANDU 6 plants. In this case it was determined that both the event's direct cause and the event's root cause were the liquid relief valve diaphragm rupture and the diaphragm degradation symptoms were not detected early on.

Besides, the operating personnel are usually trained at Gentilly 2's full scope simulator to cope with this event by using a specific abnormal event procedure. During the operative actions performed to manage this event, the training usefulness was demonstrated.

The failed liquid relief valve diaphragm was replaced by another one of non-updated design. This diaphragm had been segregated due to the fact that both the initial storage period and the extension storage period defined in the applicable procedure had expired. As there weren't new design diaphragms available as spare parts, the utility decided to use an old design diaphragm.

Lessons learned / Corrective actions:

The ARN have evaluated the above mentioned issue and required:

- To perform an urgent replacement the four liquid relief valves diaphragms by new design diaphragms and;
- To acquire and maintain a minimal stock of new design diaphragms.

In order to minimize the consequences in case of new failures occurrences till the replacement with the new design diaphragms, the plant decided to implement a number of actions:

- To increase heavy water plant inventory;
- To implement daily routine inspections to the four liquid relief valves;
- To strength operation personnel training in the use of the specific abnormal event procedure.

VIII.2.6 Primary Heat Transport System - Liquid Relief Valve Spurious Opening (CNE Feb 2nd 2006)

CNE was operating normally at full power, when the opening of the liquid relief valve (LRV) 3332 PV4 was detected. The safety systems were not demanded to actuate and this event was considered of very low safety significance. The plant was lead to cold shutdown condition according to what is established in the applicable operation procedures. CNE was 48 hours out of service.

The LRV's spurious opening was caused by the diaphragm rupture of the corresponding quick exhaust valve (QE) 6332 QE4. It was verified that the material of the failed diaphragm was Urethane instead of Vitón as is specified. Besides, the Urethane diaphragm's lifetime used had expired.

During the last scheduled outage a modification was implemented in the opening and closure time of two of the LRVs (3332 PV4 and PV13). This modification was implemented because the CNE designer (AECL), recommends that LRVs should open in less than one second (to give the primary heat transport system – PHTS - adequate protection) and close in more than three seconds (to avoid water hammer affecting the PHTS).

In order to comply with this recommendation, both a new pilot valve with more capacity to evacuate the service air (used to command the LRVs) and a QE valve to accelerate the LRVs opening were installed. Besides, to augment the LRV's closure time a restriction orifice in the air inlet to the pilot valve was installed, so the air ingress is slow and the LRVs closure is soft. These valves were qualified to resist the design base earthquake and the more relevant LOCA.

When the QE valves were replaced, due to human and procedural errors, valves with Urethane diaphragm instead of Vitón were used. The above-mentioned errors were due to the fact that the valves that were changed did not have the diaphragm material identification.

Lessons learned / Corrective actions:

The event's root cause was attributed to an inadequate elastomer management policy; faults due to ageing effect were not considered. Besides, the diaphragm installation procedure does not include a verification of the adequacy of the elastomer to be used.

The direct cause observed was the QE valve diaphragm break corresponding to the LRV 4. The corrective actions taken were:

- To change the urethane diaphragms for vitón diaphragms in the QE valves that were replaced.
- To modify the diaphragm installation procedure including an elastomer verification

VIII.2.7 Loss of Heavy Water In Steam Generator #1 (CNE Jul 19th, 2006)

ENPP was operating normally at full power. During the daily feedwater system sample routine, a high tritium concentration in the water purged (secondary system) from the steam generators was detected.

By means of gamma activity measurements in such purges, the existence of a leakage of heavy water from the PHTS to the secondary loop into the steam generator Nr. 1 was confirmed. Then the corresponding procedure to estimate the heavy water leakage rate evolution was applied. According to the mentioned procedure, the leakage rate evolution was followed till it reached 5 Kg. / hour, due to the fact that only leakage rates higher than this value allow to detect the failed tube by fluorescein test.

Finally, on July 24th it was decided to shutdown the reactor according to what is established in the applicable operation procedures, remaining out of service during almost seven days. This event was considered as of very low safety significance.

The heavy water leakage was estimated in about 463 Kg. and the activity released to the secondary system reached 4,4 E13 Bq. of Tritium and 1.1 E08 Bq. of Iodine (about 15 % of the monthly discharge derived limit).

The direct cause observed was the steam generator #1 tube failure located in the U-bend zone in X5; Y70 position. The root cause was that the scallop bars design does not fulfil the goal of tying the steam generator tubes in the U-bend zone because such scallop bars have degradation signals that enable fretting among them and the steam generator tubes, wearing out the tube wall.

VIII.2.8 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 occurred in CNA I and CNE during 2004 – 2007 are given with less details

EVENT / OPERATING EXPERIENCE	EXAMPLES OF LESSONS LEARNED / CORRECTIVE ACTIONS
Plant manual shutdown as a consequence of N04 channel closure leakage (CNA I, 17/09/05)	-Replacement for new ones all channel closure welded lips -Incorporate metallic spiral wounded gasquets instead of lips welded configuration
Manual Shutdown at CNE due to “high temperature in Primary System Pump seal” (CNE 15-12-06)	-Review all RTD’s (Resistance Temperature Detector) in Primary System Main Pumps seals after finding an inconsistency between design Technical Specifications and specifications of devices installed for measurement of seal’s temperature. -Provisional (before implementation of final corrective actions), establishment of more conservative alarm set points by pump seals high temperature. -Start of a research program to determine root causes of early aging of cable / connectors in RTD´s
Loss of D_2O through mantel in position K15 (CNA I, 17-12-06): WANO EAR PAR 07-016	-Modification of Work Plan incorporating mechanisms to guarantee The proper channel closure nut adjustment check and the addition (into the work plan) of verification in the adjustment of nuts in positions around channels, which were worked. -Qualification of personnel who retires nuts and emphasize the responsibility of task’ supervisor and the importance of the job. -Use of the event for “lessons learned”: Spread this event between operation personnel and maintenance personnel including supervision level.
Steam Generator tube leakage at CNA I (16-03-07)	-Identification of localized reduction of diameter (denting) of the failed tube -Evaluation –in course- of possible causes related with secondary system chemistry and condenser tubes material -Review of scallop bars design which not fulfil the goal of tying the steam generator tubes in the U-bend zone because such scallop bars have degradation signals that enable fretting among them and the steam generator tubes, wearing out the tubes wall

EVENT / OPERATING EXPERIENCE	EXAMPLES OF LESSONS LEARNED / CORRECTIVE ACTIONS
Moderator heat exchanger tube failure (CNA I, 21/09/06) WANO EAR PAR 07-004	-Improve the Foreign Material Exclusion: control of tasks, analyse to design devices to detect and remove foreign objects from heat exchanger -Heat Exchanger replacement analysis consult to the designer -Radiological personal doses diminished compared with the previous similar work in the same heat exchanger

ANNEX IX

RESUME OF NASA QUALITY ASSURANCE GENERAL MANUAL CONTENT

IX.1. TABLE OF CONTENTS

IX.2. QUALITY POLICY

Nucleoeléctrica Argentina S.A., 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**
  Meet the applicable standards for the different 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 fulfillment.
- **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.

IX.3 ENVIRONMENTAL POLICY

Nucleoeléctrica Argentina S.A., an organization committed to electric generation in CNA I, electric generation and Cobalt production in CNE, maintenance construction and commissioning in CNA II, Technical and Administrative Direction in 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**
  Meet the applicable standards for the different 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 fulfillment and make it available to all parties involved. To 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 Quality Management**
  Try to achieve continuous improvement by the systematic and periodical assessment of environmental management.