NATIONAL NUCLEAR SAFETY REPORT

CONTENTS

INTRODUCTION

The First National Nuclear Safety Report was presented at the first review meeting of the Nuclear Safety Convention. At that time it was concluded that Argentina met the obligations of the Convention.

This second National Nuclear Safety Report is an updated report which includes all safety aspects of the Argentinian nuclear power plants and the measures taken to enhance the safety of the plants. The present report also takes into account the observations and discussions maintained during the first review meeting. The conclusion made in the first review meeting about the compliance by Argentina of the obligations of the Convention are included as Annex I. In general, the information contained in this Report has been updated since March 31st 1998 to March 31st 2001.

Those aspects that remain unchanged were not addressed in this second report with the objective of avoiding repetitions and in order to carry out a detailed analysis considering article by article.

As a result of the above mentioned detailed analysis of all the Articles, it can be stated that the country fulfils all the obligations imposed by the Nuclear Safety Convention.

The questions and answers originated at the first review meeting are included as Annex II.

With the aim of facilitating the understanding of the Second National Nuclear Safety Report the main introductory aspects of the First Report are reproduced.

I.1 General Aspects

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

This Report describes the actions Argentine Republic hascarried out since the first nuclear safety report (March, 1998) up to March, 2001 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 and procedures, 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 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. A third nuclear power plant, Atucha II, is in an advanced state of construction.

Atucha I nuclear power plant 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 nuclear power plant 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).

Atucha II nuclear power plant is under construction, very near 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.

I.2 National Policy in the Nuclear Field

Due to its special characteristics, the activities related to the use of nuclear energy for peaceful purposes need 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 National Constitution.

Within this context, Act No 24,804, 1997 or "National Law of the Nuclear Activity", is the legal framework for the peaceful uses of nuclear energy. Article 1st of the Act No 24,804, 1997, sets 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 and the Nuclear Regulatory Authority.

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

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

I.3 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 Safetyand the most significant conclusions introduced during the first review meeting in 1999. This means that the Report has been ordered according to the Articles of the Convention on Nuclear Safety and the contents indicated in the above mentioned Guidelines.

Article 6 describes the actions adopted by the organization in charge of the operation of the nuclear power plants, in order to evaluate or improve safety; such actions result as a consequence of operational experience, backfitting measures or in response to regulatory requirements.

Article 7 presents the changes about the legal and regulatory structure that rules nuclear safety. It also includes the changes on the normative framework.

Article 8 presents the minor changes concerning functions, responsibilities and organizational structure of the Regulatory Body, human and economical resources. It also includes the Quality System applied in the Regulatory Body.

Article 9 deals with the regulatory controls of the fulfillment of the Licensee's responsibilities.

Article 10 highlights aspects related to safety culture.

Article 11 deals with functions, responsibilities and the structure of the organization in charge of the operation of the nuclear power plants, human and economical resources and personnel qualification. It also includes aspects about deregulation and privatization of the electric market.

Article 12 analyses the methods to prevent, detect and correct the occurrence of events related with human factors.

Article 13 updates the activities of the quality assurance program and quality assurance manual. It also includes the updated nuclear power plants organization charts.

Article 14 deals with the results of the ageing program applied to the nuclear power plants and the Regulatory Plant Safety Indicators.

Article 15 shows the new values of radioactive release into the environment, public and occupational exposures, individual and collective doses.

Article 16 presents the changes about the laws, regulations and requirements existing in the country and their implementation in case of a radiological emergency in a nuclear power plant.

Article 17 presents the site re-evaluation studies.

Article 18 analyses issues concerning the construction of new nuclear power plants.

Article 19 presents the results of the operation experience feedback program. It also includes aspects related to accident management and severe accidents and the activities between the Responsible Organization and WANO.

1.4 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 of the obligations of the Convention.

Annex II includes the questions and answers discussed at the first review meeting .

ARTICLE 6
EXISTING NUCLEAR INSTALLATIONS

In the First National Nuclear Safety Report (NNSR) some improvements to the nuclear power plant safety have been detailed. Particularly, the major significant requirements established by the Regulatory Body have been stated for both Atucha I and Embalse nuclear power plants.

In this Article the state of those requirements and the new safety aspects taken into account since the first review meeting up to the present report are presented (see Annex I). Special attention was paid to the new and extended backfitting program for Atucha I nuclear power plant based on the results of the continuous safety assessments and the Regulatory Body requirements.

6.1 Safety Assessments and Corrective Actions in Nuclear Installations

6.1.1 Atucha I Nuclear Power Plant - Plant Improvement Status

6.1.1.1 Coolant channel replacement

As it has been mentioned in the first report, reactor internals were originally designed considering the reactor lifetime. However, the coolant channels have "Stellite-6" (a 60% cobalt alloy) located in two contact areas. Therefore, a coolant channel replacement program was established by the Regulatory Body as part of the plant backfitting program (see 6.2.1.3 NNSR 1998). The work has been in progress since then and the remaining 54 (approximately 80% of the total coolant channels were replaced) coolant channels containing "stellite-6" have already been scheduled to be replaced.

During coolant channel replacement activities it was realised the existence of coolant channel nozzle stuck in their guide bushing, due to effects such as crud and fuel channel thermal isolation (foils 0.1-0.2 mm of Zircalloy 4) particles deposition in the clearance. Besides, it was determined that coolant channel axial growth due to irradiation is another effect that contributes to the coolant channel sticking against the moderator tank bottom. In order to keep into account the mentioned effects it was decided to implement a channel design modification that consists on a larger clearance, a cover jacket and a spring adjuster (see 6.1.1.3).

6.1.1.2 Coolant channel thermal isolation replacement

Operative experience demonstrated that coolant channel thermal isolation suffered from an increasing degradation due to creep, deformation and embrittlement causing foil fractures. This process has generated a relevant quantity of particles which are disseminated into the moderator tank.

Last years inspections and channel replacement activities showed that the damage of the foils was significant and increasing. Additionally, foil particles have been found within the steam generator's tubes, moderator's heat exchangers tubes and, upper and lower reactor vessel plenum.

The Regulatory Body made requirements to have coolant channels with foil 0.4 mm instead of 0.1-0.2 mm, as the original channels, in order to avoid the foil debris generation up to the end of the reactor lifetime.

6.1.1.3 Extended coolant channel replacement program

The Regulatory Body required in 1999 to extend the coolant channel replacement program to take into account the channel issues (see 6.1.1.1 and 6.1.1.2) and to fulfil the following main objectives:

- ✓ Moderator tank should be clean (without foil debris);
- ✓ Coolant channels and tube guides nozzles should have a free axial movement;
- ✓ Coolant channels thermal isolation (foil) should maintain its integrity during all plant operation lifetime and
- ✓ Coolant channels should not contain "Stellite-6" alloy.

The Responsible Organization put in practice this regulatory requirement implementing a coolant channel design modification that considers measures to solve the nozzle sticking problem, the irradiation growth and the foil creep effect. A new coolant channel replacement program is in progress considering such problems.

Besides, once the coolant channel replacement program finishes, the Responsible Organization will follow the coolant channels performance through an improved surveillance program.

6.1.1.4 Moderator heat exchangers

Atucha I nuclear power plant was designed with two moderator heat exchangers, each of 100% capacity. Maintenance and inspections activities of the moderator heat exchangers were not considered in the original design. However, corrective actions were necessary to be performed recently in the moderator heat exchanger # 2 due to a heavy water tube leakage.

The surrounding area of moderator heat exchanger is a high radiation zone, hence significant efforts were necessary in order to carry out such maintenance activities in terms of the number of personnel involved and the development of special tools.

Besides, so as to develop this new activity an intensive personnel training was required. Therefore, the ALARA principle was applied in the design of the special tools such as cutter devices remotely operated, welding machines and remote handling devices (master slave mechanisms).

6.1.1.5 Second heat sink

As it was stated in the first report, a second heat sink representing an additional heat sink using feed and bleed via the steam generators has been envisaged. This heat sink system has the main purpose of ensuring, as an additional emergency system, the heat transfer via the steam generators in cases where the normal heat sink or the high pressure residual heat removal system are either unavailable or ineffective.

The detailed engineering of this new system was already completed and it is now being constructed as it was foreseen in 6.2.1.4 NNSR 1998. The schedule has been adapted to the planned outages required to carry out the coolant channel replacement program.

Additionally, the Second Heat Sink is then going to function as:

- ✓ Emergency Feedwater System

  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 diesel machines.

- ✓ Main Steam Relief Station

  A two separate, independent and redundant control and gate valves with the necessary I&C equipment with the capability to control the steam pressure.

6.1.1.6 Plant specific probabilistic safety assessment

Within the framework of probabilistic safety assessment program and as an extension of Atucha I PSA Level 1 (6.2.1.6 NNSR 1998), the internal fire analysis and safety analysis implying other sources of radioactive release, different from the reactor core, were concluded. Besides, the first phase of shutdown probabilistic safety assessment (middle loop operation) required by the Regulatory Body was also finished.

6.1.1.6.1 Fire risk analysis

The study was developed in two stages:

1. The first stage was a screening analysis of all plant areas that have safety-related components in order to select, in a conservative way, the areas that contribute to core damage frequency. The analysis was made with the assumption that once fire starts in a given area, all the equipments and cables in the area will become unavailable immediately.

2. The second stage was a detailed analysis of each selected area in the previous stage. The estimated failure of different components and systems is dependent on the postulated scenarios by using probabilistic and deterministic tools for the propagation phase of the fire.

There is an important contribution to the core damage frequency due to fire from those scenarios that lead to:

- ✓ Loss of all heat sinks.
- ✓ Loss of all instruments and plant control.
- ✓ Loss of plant control under evacuation of the control room.
- ✓ Loss of D.C. electric bus bars.

Electrical and electronic cabinets location are the most significant contributors to core damage frequency. The area of electrical boards corresponding to the low voltage emergency distribution system contributes with a 13% to core damage frequency of fire. The area corresponding to the control room contributes in a 9% to core damage frequency of fire.

6.1.1.6.2 Accident analysis addressing radioactive releases different from reactor core.

The failure of gaseous, liquid and solid waste management systems, spent fuel elements storage system and spent fuel elements transport system were included within the analysis.

The gaseous waste management system evaluation showed that releases to the environment, in the worst case would exceed 30% the daily discharge limit (as a reference value) and its occurrence frequency is below 10^{-6} / year. The liquid waste management system evaluation showed that the daily discharge limit will not be exceeded in any case.

From the analysis of the fuel handling and storage systems the events that need to be highlighted are those related to the mechanical damage of fuel elements into the spent pool storage. These events combined with the activity detectors failures located into the water discharge line would produce a radioactive release that would not exceed the annual value of the gaseous releases allowed limits in normal operation as a reference value.

From the spent fuel elements transport system, the highlighted events are those related to hose breakages and failures that cause the loss of the spent fuel element coolant. The hose breakage is supposed to be a relatively frequent event due to the fact that it is considered a single failure.

6.1.1.6.3 Probabilistic Safety Assessment during shutdown

As part of the study “Atucha I - PSA for Low Power and Shutdown Modes" a first stage, corresponding to cold depressurised shutdown state, was completed. Within this plant condition two operating states are identified corresponding to the availability of heat removal systems. Such operating states are time dependent on the reactor shutdown. In the second state, in addition to the residual heat removal system, there is another core heat removal system. (Regulation and Control Volume System).

Shutdown activities as well as availability of particular components are taken from a selected reference scheduled shutdown.

Initiating events identified are essentially of three types:

- ✓ loss of power to essential emergency loads,
- ✓ failures of particular residual heat removal equipment,
- ✓ loss of ultimate heat sink.

Main contributors to core damage frequency come from a spurious signal that isolates components in the heat removal system without a direct fault indication (failure in diagnosis). The relatively high maintenance unavailability of certain components has an important weight, there is also an important contribution of single failures that cause the unavailability of components that can be used as redundant of residual heat removal system.

6.1.1.7 Pressure vessel integrity

The Regulatory Body required the Responsible Organization to take the necessary actions to heat the water contained in the high pressure accumulators of the emergency core cooling system if it can not be demonstrated that the reactor pressure vessel integrity will continue preserving an appropriate safety margin at the end of the reactor lifetime (6.2.1.7 NNSR 1998).

As a consequence of such requirement the Responsible Organization performed a pressurized thermal shock analysis. The fracture mechanics analysis was based on the results of the detailed thermal-hydraulics evaluation. The safety analysis was carried out in two steps. The first step was the selection of the most severe transient, the second step was the full elastic-plastic analysis of the selected transients.

The analysis showed an increase in the safety margin defined as the difference between temperature for ductile-to-brittle transition at the end of CNA I lifetime and the admissible limit temperature. The safety margin was increased from 1°C to 15°C. This result shows a better reactor pressure vessel safety margin at the end of lifetime leading the Responsible Organization to ask for exemption to heat up the water contained in the high pressure accumulators.

At present, the Regulatory Body is evaluating the pressurized thermal shock analysis in order to assure that an adequate reactor pressure vessel reliability at the end of the life time will be maintained.

6.1.1.8 Safety assessment due to the introduction of slightly enriched fuel elements

The transition from the original fuel elements to equilibrium core with slightly enriched fuel elements was concluded (see 6.2.1.8 NNSR 1998). The corresponding safety assessments of such core design modification and the follow up of the reloading strategy for such transition were reviewed by the Regulatory Body.

Assessment of the effects of spatial xenon oscillations.

Spatial Xenon oscillations condition, particularly in the axial direction are likely, due to design features of the reactor core of CNA I. The use of partial insertion rods allow the attenuation of such oscillations. During start up it was noticed that the oscillations were so small in amplitude that they were even undetectable, so that the use of partial insertion rods was not necessary while the plant operated with natural uranium fuel elements.

As the existing fuel elements were gradually replaced by slightly enriched uranium dioxide fuel elements, the oscillation amplitude became more significant under normal operating conditions.

Several calculations were carried out in order to evaluate xenon oscillations, particularly during a cycle 100%-80%-100% full power, and the results obtained were compared with those coming from actual cycles, having different numbers of slightly enriched fuel elements. The agreement between calculation and measurements was satisfactory. The oscillations observed were always attenuated.

An event of increase in the oscillation amplitude was observed, which was due to fuel reloading and power reduction that induced control rod movements strengthening xenon oscillations. All the operation limits were preserved; alarms and compliance with operation instructions were as expected and effective.

Nevertheless, deficiency in predicting the oscillation as a function of refuelling and/or control rod movement, indicated that it was necessary to improve operation procedures in order to prevent oscillations increasing amplitude, particularly for the case of intensive refuelling. The following corrective measures were required:

- ✓ to modify operation procedures in order to avoid fuel reloading or any other operation which could generate control rod movements in phase with the oscillation.
- ✓ to use partial insertion control rods originally designed for the attenuation of oscillations, when such oscillations are beginning to be detected by nuclear instrumentation.

These corrective measures were implemented satisfactorily, and their goal was achieved.

6.1.1.9 Shutoff control rods

Last years there was a slight but constant increase in the drop time of some shutoff control rods. Therefore, the Regulatory Body required the Responsible Organization a close follow up of the shutoff control rods performance.

In 1999 scheduled outage some nozzles of tube guide control rods were stuck in their guide bushing, due to both effects, crud and fuel channel thermal isolation particles deposition in the clearance as it was the case of the coolant channels (see 6.1.1.1). Besides, the axial growth of tube guide control rods by irradiation caused it to stick against the moderator tank bottom, similarly to coolant channels

With the purpose of evaluating the actual performance, several tests, inspections and analysis were required by the Regulatory Body and, among them, the following can be mentioned:

- ✓ Drop test performance from several rod positions.
- ✓ Inspection program focused on detecting any eventual interactions between control rod guide tubes and coolant channels.

- ✓ Evaluations of the accident analysis that imply requirements of the shutdown system, present in both probabilistic safety assessment (PSA) and Final Safety Analysis Report.

As a conclusion of the mentioned evaluation it was observed that the design of the shutdown system is very conservative in terms of the reactivity shutdown margin, as well as the number of necessary rods for the reactor safe shutdown.

A number of shutoff control rods and tube guides were required by the Regulatory Body to be removed from the core to carry out several tests and inspections, neutron fluence calculation, metalographic analysis; fractomechanical and brittleness/ductility evaluations. A lifetime evaluation of the control rods was also required.

Based on the evaluation results of the shutoff actual performance, the Regulatory Body requested the Responsible Organization an upgraded program to replace all the tube guide control rods.

6.1.1.10 Backfitting program

On November 1999, considering the above items and requirements, the Regulatory Body issued an updated backfitting program aimed at priorizing the reactor internals issues. The new program also considers relevant safety aspects such as the completion of the second heat sink system (6.1.1.5) and the reactor pressure vessel analysis (6.1.1.7). Besides, an updated version of the relevant safety documentation was required.

The main activities included within the backfitting program are:

- ✓ A control rod shutdown system test program with the objective of detecting early effects of potential failures in control rods.
- ✓ Replacement of all channels with “Stellite-6”.
- ✓ Replacement of all control rod guide tubes with a new nozzle design.
- ✓ Updating of the Safety Report, the Probabilistic Safety Assessment, Operation Policies and Principles Manual, Maintenance Manual and Quality Assurance Manual.
- ✓ In-core neutron flux sensor guide tubes replacement.
- ✓ Moderator tank cleaning.
- ✓ Commissioning of the second heat sink system.
- ✓ Pressure vessel integrity analysis.

The backfitting activities required by years 2000 and 2001 were completed as stated by the Regulatory Body. The remaining tasks to conclude the backfitting program will be carried out by 2002.

6.2. Embalse Nuclear Power Plant

6.2.1 Pressure tube inspection

Garter springs repositioning program

In 1998, Atomic Energy Canada Limited completed a pressure tube blister susceptibility assessment to evaluate the potential for hydride blister formation in the pressure tubes of Embalse nuclear power plant (see 6.2.2.2. NNSR 1998). Several deuterium ingress cases were analysed using deuterium concentration data from other CANDU 6, concluding that for each case pressure tubes could potentially reach blister threshold formation. Therefore, it was required to complete a long-term assessment of hydride blister susceptibility by monitoring deuterium uptake in pressure tubes at Embalse.

Regarding the above mentioned considerations, the Regulatory Body required the following:

1. To demonstrate that there are no pressure tubes containing Hydrogen/Deuterium over blister threshold formation using a specific plant model to determine deuterium content and uptake. The model had been adjusted from scrapping results performed during 1998 planned outage.

2. To submit a new garter springs repositioning program for remaining pressure tubes, according to model results.

The utility fulfilled the requirement as follows:

During 1998 Embalse nuclear power plant planned outage it had been implemented an in-core scrape sampling program, which involved a total of 10 pressure tubes that were sampled after about 109,600 equivalent full power hours. The purpose of pressure tubes scrape sampling was to monitor the uptake of deuterium. The samples were selected by the Responsible Organization staff from a list of recommended candidate channels issued by Atomic Energy Canada Limited to provide an indication of the variation in deuterium concentration along the length of the pressure tubes.

Scrape sampling was performed using the wet scrape tooling in which a separate tool is delivered to each sampling location in sequence by the fuelling machines and cutting is performed fully immersed in the primary heat transport system coolant under shutdown temperature and flow conditions.

The sample analysis were performed by using two techniques. The non destructive measurement of the temperature corresponding to the Terminal Solid Solubility for hydride Dissolution using Differential Scanning Calorimetry and Hot Vacuum Extraction – Mass Spectrometry. Differential Scanning Calorimetry is normally performed only to provide consistency check on the Hot Vacuum Extraction – Mass Spectrometry results.

The assessment of Embalse nuclear power plant scrape sampling results, performed in 1998, indicated lower deuterium uptake rates in comparison with previous assessment performed using deuterium concentration data from other CANDU 6 reactors.

The assessment of Embalse nuclear power plant scrape sampling results, determine that until the reactor reaches 160,800 EFPH, that means in October 2004 approximately, conditions to blisters formation would not be given.

A set of (ultrasonic and eddy current) measurement techniques of the interesting parameters as well as the “Spacer Location and Repositioning” program, have been applied in Embalse nuclear power plant since the late eighties. Today this program is under execution, having been inspected 201 and repositioned 195 channels out of 380.

6.2.2 Feeders

An important task about CANDU Reactors is the feeders inspection based on the operating experiences in other CANDU reactors. During the last planned outage several feeders areas were inspected, with the objective of determining the evolution of the thickness wall decrease by erosion – corrosion mechanisms. The inspection methods and qualified personnel used are in accordance with the standards and guides. The results in the evolution of the inspection that took place allow a life expectancy higher than 25 years for all sensitive areas.

6.2.3 Dry storage of irradiated fuel elements

At present the utility has programmed to extend the Dry Storage of Irradiated Fuel Elements System (see 6.2.2.4. NNSR 1998) by constructing new silos to cover plant operation needs for the next five years. The program establishes the construction of forty new Silos during year 2001.

New silos´ design, construction and assembly do not foresee changes from the existing Silos, the safety related issues fulfil the safety requirements as the existing Dry Storage of Irradiated Fuel Elements System licensed.

6.2.4 Plant specific Probabilistic safety assessment.

The followings tasks have been performed:

- ✓ Identification and grouping of Initiating Events;
- ✓ Event Trees modelling;
- ✓ Preparation of the Analyst documents;
- ✓ Failure Mode and Effect Analysis (FMEA);
- ✓ Fault Trees modelling,
- ✓ Plant significant events analysis and;
- ✓ Header definitions and event sequences modelling.

During the PSA development, advantage has been taken from the knowledge gained leading to: preventive maintenance improvements, design modifications, procedure modifications, auxiliary water pump functional tests under extreme conditions, existing plant documentation improvements and improvement to the models used to calculate safety systems annual performance indicators.

At present, the final PSA quantification is not concluded. Therefore, the improvements and modifications impact will be evaluated when model quantification is finished.

6.3 Atucha II Nuclear Power Plant

The plant construction activities are still discontinued, keeping the maintenance of the components meanwhile decisions are expected to be taken so as to continue with the construction of the plant.

Nevertheless, during last three years some activities have to be highlighted:

Assembling of the pressure vessel, the four moderator pumps, 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 rods switches for its own consumption.

Commissioning of the reactor Building and Turbine Building crane.

Components Storage (for example main turbine) in the rooms where they will be assembled, when they were almost conditioned to normal operation.

Regulatory Activities.

Since 2001, considering that the construction might be continued, the Regulatory Body increased its regulatory activity, mainly concerning the following tasks:

1. Verification of the impact of CNA I operative experience to the design of CNA II.
2. Verification of the maintenance of the components stored within the site, the ones that are already installed or the ones that are in service.
3. Update of the applicable Regulatory Body Standards, taking into account safety evolution since 1979 when CNA II construction was decided.
4. Making use of Brazilian Regulatory Authority experience in Licensing Angra II Nuclear Power Plant, taking into account that the designer authority is the same as for Atucha II nuclear power plant and that its construction was also stopped for a long period.

6.4 Opinion of the Regulatory Body Concerning the Operation Continuity of Nuclear Installations

CNA I and CNE fulfil the regulatory standards related to design and operation. In the case of CNA I, the Regulatory Body conditioned the operating license to the fulfilment of the backfitting program detailed before, presently being carried out.

ARTICLE 7
LEGISLATIVE AND REGULATORY FRAMEWORK

7.1 National Legislative Framework

There have not been major changes within the Legislative and Regulatory Framework described in the first national nuclear safety report. However, a change that can be mentioned is related to Decree 1390 dated November 1998 which established that the Regulatory Body will be in charge of approving the contingency plans in case of nuclear accidents, programs to face emergencies, and in case when it is necessary, of the training of workers and public in general (see Article 16). Additional functions to the Regulatory Body related with the National Waste Management Regime, as it was set in Act No 25,018 (Article 8), have been established.

7.2 Normative Framework

The Regulatory Body has continued updating the regulations in force, particularly modifications have been made in the following standards related to the safe operation of nuclear power plants:

Table 7.1 – Updated Standards during 1999 - 2001

AR Code	Name
0.11.1	Conditions to Obtain Specific Authorizations for Personnel of Major Installations
3.3.1	Reactor Core
3.3.2	Heat Removal Systems
10.16.1	Transport of Radioactive Materials

Table 7.2 New standards included during 1999 - 2000

AR Code	Name
0.11.3	Training of the Licensed Plant Personnel
10.12.1	Radioactive Waste Management

7.3 Licensing System

Nuclear power plants personnel licensing

Power plants personnel licensing system has remained without changes. On the other hand, continuing the updating process of the normative system, a new Standard AR 0.11.3 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.

ARTICLE 8
REGULATORY BODY

8.1 Functions and Competence of Regulatory Body

As it was stated in the first National Nuclear Safety Report Act No 24,804, 1997 or "National Law of Nuclear Activity", sets that the Nuclear Regulatory Authority (Regulatory Body) is in charge of the regulation and surveillance of nuclear activity concerning radiological and nuclear safety, physical protection and safeguards. On Sept. 1998 Act 25,018 -Radioactive Waste Management Regime- came into force. This Law sets requisites that involve the Nuclear Regulatory Authority in the management of Radioactive Wastes. Articles 7º, 8º and 9º set that the Nuclear Regulatory Authority shall:

- ✓ Approve the acceptance criteria and the transference conditions of the radioactive waste formulated by the National Atomic Energy Commission (Act No 25,018 application authority).
- ✓ Approve radioactive waste transference procedures, in particular irradiated fuel elements established by the National Atomic Energy Commission (the Commission, in this case as the radioactive waste producer centre).
- ✓ Advise the National Congress in relation to the Radioactive Waste Management Strategical Plan, made and updated every three years by the National Atomic Energy Commission.

It must be highlighted that part of the functions that Law attributes to the Nuclear Regulatory Authority within the Waste Management Area were "de facto" already being executed as a part of the Regulatory Authority action plan.

8.2 Regulatory Body Organizational Structure and Human Resources

On March 2000, after new national authorities had been elected, the Board of Directors of the Nuclear Regulatory Authority has been renewed. The Institution still reported to the Argentine Presidency during 2000, through the Productive Innovation, Science and Technology Secretary. Since February 2001 the Nuclear Regulatory Authority reports directly to the Secretary of the National Presidency as it had been before year 2000. The organizative structure and organization chart remained unchanged.

The Nuclear Regulatory Authority has 202[1]people. The administrative staff have been reduced in 19 people due to volunteer retirement proposed by the national authorities. This reduction has had a minor impact on human resources applied to the regulatory control of nuclear power plants. Although these people were not involved into tasks related to the regulatory control of the nuclear power plants, their tasks have been distributed among the Regulatory Body remaining staff.

8.2.1. Resources assigned to the nuclear power plants regulatory control

8.2.1.1. During Operation

Tables 8.1 and 8.2 show, respectively, the direct and indirect distribution of human resources dedicated to power plants regulation, updated during 2000. The comparison between the values (in man-days) corresponding to 1998 and 2000 show that:

- ✓ Human resources (total, direct and indirect) have diminished in a 6% if it is compared 1998 to 2000.
- ✓ The same comparison shows that while the direct human resources have decreased, the indirect ones have increased.

This is due to the fact that several significant projects have finished and that the methodologies used for the application of human resources to activities and projects have been improved, obtaining much realistic estimations.

Table 8.1 - Human Resources Dedicated to Projects and Activities Directly Related to Nuclear Power Plant Safety

PROJECTS AND ACTIVITIES	MAN-DAYS / YEAR (2000)
Inspections and evaluations in power reactors	2695
Severe Accidents in Nuclear Power Plants	605
Incident Reporting System	110
Core material behaviour	176
Analysis of Atucha I behaviour during severe accidents in early phase by means of mechanists methods	121
Safety Evaluations based on Thermohidraulic Codes	440
Software reliability	176
Regulatory use of the probabilistic safety assessment	231
Research and application of methodologies for the analysis of Nuclear Power Plants incidents	154
Evaluation of the action of external events in Atucha and Embalse sites	55
TOTAL	**4763**

(1) The Nuclear Regulatory Authority is made up of:

Real:	197
Reserved vacancies:	5
Total:	202

The total staff does not include the grants or hired staff:

Grants:	18
Hired:	13
Total	31

Table 8.2 – Human Resources Dedicated to Projects and Activities Indirectly Related to Nuclear Power Plant Safety

PROJECTS AND ACTIVITIES	MAN-DAYS / YEAR (2000)
Dosimetry projects	512
Radiation measurements	31
Environmental studies	759
Remote surveillance in nuclear power plants	506
Dose records	238
Medical response and dosimetric evaluations in case of accidental or occupational exposures	739
General administration	1275
TOTAL	4060

8.2.1.2. During construction and commissioning

During June 2000 a working group was constituted with the aim of reinitiating the regulatory studies related to Atucha II Power Plant Project. This group was made up of 6 people, with an average dedication of 57%.

8.3. Financial Resources

The evolution of the amount of expenses estimated and actually spent during 1998 through 2000 show that even though there have been some restrictions, real expenses have not been reduced substantially. On the other hand, the mentioned restrictions have not affected regulatory activities in relation to power plants or any other relevant installation but the services area.

Table 8.3 – Evolution of the Expenses Estimated and Spent during 1998 through 2000

(in U$S)	1998	1999	2000
Estimated expenses	18.498.445	17.882.681	19.020.154
Real expenses	17.417.654	16.633.401	16.132.828

8.4. Political Changes that Influenced in the Implementation of the National Systems

The Nuclear Regulatory Authority, as an autarchic entity, keeps on performing its functions within the frame of the Argentine regulatory system with absolute normality. The Board renewal that took place in 1999 did not alter politics or the actions taken by the Nuclear Regulatory Authority. The Institution keeps the entire support of the governmental authorities.

8.5. Quality Systems in Regulatory Authorities

The Nuclear Regulatory Authority, as a “de facto” and “de jure” independent authority, keeps on taking actions with the objective of improving the quality of its regulatory performance since a long time ago. The following achievements have been accomplished:

- ✓ An annual workplan and its management control, as well as the permanent development of technical and administrative procedures applicable to regulatory activities.
- ✓ Continuous improving of the existing normative system by means of writing new standards and revising the existing ones.
- ✓ Technicians and Professional annual training program.
- ✓ Federal jurisdiction for nuclear regulatory activities. This process was emphasized with the opening of the South Regional Office and will go on with the creation of Delegations in other country regions.
- ✓ Public release of the Nuclear Regulatory Body Annual Report, sent to the National Congress and the publication of the technical and scientific works.

A plan to improve quality within the Organization that includes immediate actions was established. Among the immediate actions the following might be mentioned:

- ✓ Strengthening of the regulatory attitude by means of training and making the staff aware.
- ✓ Internal audits.
- ✓ Improving the management.
- ✓ Management training.
- ✓ Gradual implementation of Regional Offices.

8.6. Relationship with other Organizations

Relationship between the Nuclear Regulatory Body and other organizations remains the same as far as regulatory activities are concerned.

The participation in the Forum of Ibero-American Nuclear Regulators and the Network of Regulators of Countries with Small Nuclear Programs goes on with no changes. The following chart shows the places and dates where the meetings took place:

Forum of Ibero-American Nuclear Regulators		Network of Regulators of Countries with Small Nuclear Programs	
Mexico	July 1997	Austria	September 1998
Argentina	May1998	Argentina	October 1999
Spain	April 1999	Finland	September 2000
Brazil	November 1999		
Cuba	July 2000		
Argentina	May 2001		

On the other hand, new technical and scientific cooperation agreements between the Nuclear Regulatory Body and other national and international organizations (tables 8.6.1 and 8.6.2) have been subscribed.

Table 8.6.1 – National Organizations

Agreement Counterparts	Agreement Objectives
Cooperation Agreement between the Nuclear Regulatory Body and Neuquen Municipality	Cooperate on the subjects that are their responsability
Cooperation Agreement between the Nuclear Regulatory Body and Navy Research Institute	Cooperate in the assistance to people accidentally overexposed to ionising radiation
Cooperation Agreement between the Nuclear Regulatory Body and the National Atomic Energy Commission	Studies on radio medicine and dosimetry methods
Cooperation Agreement between the Nuclear Regulatory Body and the National University of Buenos Aires– School of Engineering	Performing studies, advising, research and technological developments in the area of radiation protection and nuclear safety
Cooperation Agreement between the Nuclear Regulatory Body and the Argentine Geological Services	Seismic evaluations
Cooperation Agreement between the Nuclear Regulatory Body and Technical Education School "Otto Krause".	Cooperate on the subjects that are their responsability
School of Engineering students under the ARN supervision	Scientific and technological training in the area of radiological and nuclear safety, safeguards and physical protection
Supplementary Agreement between the Regulatory Body and the Gendarmerie	Installation of two aerosols monitoring stations

Table 8.6.2 – International Organizations

Agreement Counterparts	Agreement Objectives
Cooperation Agreement between the Nuclear Regulatory Body and Gesellschaft Fur Anlagen Reaktorsicherheit (GRS) about Cooperation and Exchange of information in the field of Nuclear Safety.	Cooperation and Exchange in the field of information on Nuclear Safety.
Cooperation Agreement between the Nuclear Regulatory Body and the Agenzia Nazionale per la Protezione Dell`Ambiente (AMPA), for the exchange of technical information and cooperation about nuclear safety and radiological protection.	Cooperation and experience exchange about nuclear safety and radiological protection.

And some others have been discontinued (Tables 8.6.3 and 8.6.4):

Table 8.6.3 – National Organizations

Agreement Counterparts	Agreement Objectives
Special Agreement between the Nuclear Regulatory Body and **INVAP S.E.**	Licensing of the installations that INVAP S.E. is constructing in Egypt.

Table 8.6.4 – International Organizations

Agreement Counterparts	Agreement Objectives
Agreement between the Nuclear Regulatory Body and the Electric Power Research Institute (EPRI - USA) about cooperation in the area of software reliability.	Reliability, verification, validation and licensing of Nuclear Power Plants safety programs.

Besides, specialists from the Nuclear Regulatory Body participate -as nominated experts- in the following programs and international committees:

- ✓ Waste Assessment Safety Standards Advisory Committee “WASSC” (IAEA).
- ✓ Transport Safety Standards Advisory Committee “TRANSSC” (IAEA).
- ✓ Radiological Safety Standards Advisory Committee “RASSC” (IAEA).
- ✓ Nuclear Safety Standards Advisory Committee “NUSSC” (IAEA).
- ✓ Safeguards “SAGSI” (IAEA).
- ✓ United Nations Scientific Committee on the Efects of Atomic Radiation “UNSCEAR” (United Nations Organization).
- ✓ International Commission of Radiological Protection “ICRP”.
- ✓ International Nuclear Safety Advisory Group “INSAG”.
- ✓ Argentine-Brazilian Nuclear Politics Permanent Committee.

ARTICLE 9
RESPONSABILITY OF THE LICENSEE

9.1 Regulatory Control on the Fulfilment of the Licensee`s Responsabilities

The responsibility of the Licensee was described in detail in the First National Nuclear Safety Report and there have not been any changes about this Article. However, in connection with the regulatory control of the licensee´s responsibility the following actions were taken:

On October 1999, the Nuclear Regulatory Body required the Responsible Organization the implementation of the backfitting program for Atucha I nuclear power plant to be completed during three planned shutdowns in 2000, 2001 and 2002. The implementation of this program will allow a significant improvement in nuclear and radiological safety.

On October 2000, the Responsible Organization informed the Regulatory Body about the impossibility of meeting the deadlines that had been set by the Nuclear Regulatory Body and presented an alternative program. The alternative program presented by the Responsible Organization did not satisfy the Regulatory Body. Therefore the Regulatory Body required to shutdown the plant in order to fulfil what had been originally required. The installation was shutdown in November 2000 and it was authorized by the Regulatory Body to restart in June 2001, after the completion of all the tasks required that meant a significant portion of the backfitting program. (see 6.1.1.10).

ARTICLE 10
PRIORITY TO SAFETY

The principles and priorities regarding safety were explained in the First National Nuclear Safety Report. Such principles and priorities remain unchanged since the First review meeting.

10.1 Safety Culture

Special attention was given to Safety Culture, promoted from the headquarters of the Responsible Organization 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 safety culture principles to all activities carried out at nuclear power plants.

The Regulatory Body in the case of the regulatory control principle, and the Responsible Organization in the case of the responsibility for safety principles are continuously involved in the compliance with the above mentioned principles.

Additional attitudes contributing to prioritise the safety carried out by both Responsible Organization and Regulatory Body can be mentioned:

- ✓ Independently of the personnel responsibility, the evaluation of the safety culture of the operating organization was also included in the program for renewal of personnel licenses.
- ✓ Review the safety culture attitudes during inspections by regulatory safety specialists.
- ✓ Review of trends in event reports, corrective actions effectiveness and measures implemented to prevent safety problems.
- ✓ Review of trends for special set of regulatory safety indicators.
- ✓ Assessment of minor events reporting to detect organizational weaknesses and inadequacies responses by operators.
- ✓ Increasing use of Probabilistic Safety Assessment for plant safety management.
- ✓ Improvement of relationship between the Regulatory Body and the Responsible Organization.

Efforts to improve the safety of the nuclear power plants have the highest priority at both the Regulatory Body and the Responsible Organization.

ARTICLE 11
HUMAN AND ECONOMIC RESOURCES OF THE LICENSEE

11.1 Introduction

Since the creation of NASA till the renewal of the national authorities in 1999, the instructions about the transference of Nuclear Power Plants to the private sector were kept.

In addition to the difficulties that would arise because of such transference it was found a lack of interest in the area of electric generation on the investors´ side. This lack of interest was due to the fact that the fees in the electric market did not provide an incentive to private investments.

During the transition period of privatization process, the Company took the minimum actions within the area of human resources to allow the private owner of the Power plants to apply their own human resources policy.

During 2000, after the renewal of the National Government Authorities, the process of turning NASA into a private Company was suspended and it was decided to allow the admission of young employees to lower down the average age of the working population and use the vacancies that had been produced by means of the vegetative decrease and the application of volunteer retirement plans. Thirty professionals and technicians were incorporated and they have been assigned to the nuclear power plants and the administration area staff after a training course. These personnel have been selected trying to cover a complete technical specialties areas. Besides, 130 employees that had been hired through service companies up to that moment were incorporated.

Regarding the Company Organization an only change has taken place: the Human Resources Division that reported to the Administration Management, has changed to the rank of Department and reports to the General Manager. At present this change also responds to a necessity to modify the work and salary regime, which remains unchanged till NASA separated from the National Atomic Energy Commission in 1994.

This regime, which is in process of change, has been designed to meet the needs and demands of a research and development organization and not of an electric power generating company.

11.2 Nucleoelectrica´s Staff

The total number of employees (April 2001) of Nucleoeléctrica Argentina reaches 1313 people, 298 of which are professionals, 862 are technicians and 153 are administratives.

PERSONNEL

	CNA I	CNA II	CNE	Main Branch	Total
Professionals	103	31	87	77	298
Technicians	288	137	405	32	862
Administratives	39	3	70	41	153
Total	**430**	**171**	**562**	**150**	**1313**

11.3 Economical Resources

The Company's performance in 1998, 1999 and 2000 is shown in the following chart:

PERFORMANCE

	1998	1999	2000
Gross Energy (MWh)	7.452.828	7.105.976	6.177.090
Charge / load factor (%)	84,65	80,71	70,0
Installed Nuclear Power (%)	5,4	5,0	5,0
Generated Nuclear Power (%)	10,47	9,33	7,47

During 2000, a considerable decrease in the generation was noticed. It was due to two main reasons:

- ✓ The corrective maintenance of a moderator heat exchanger in Atucha I which repair took three months.
- ✓ Regulatory enforcement to fulfill the backfitting requirement.

Low values of the tariff within the wholesale market also impacted on the incomes.

ENERGY FEES (ANNUAL AVERAGE IN $/MWh)

Year	1998	1999	2000
Value	20,53	22,42	24,1

11.4 Current Expenses

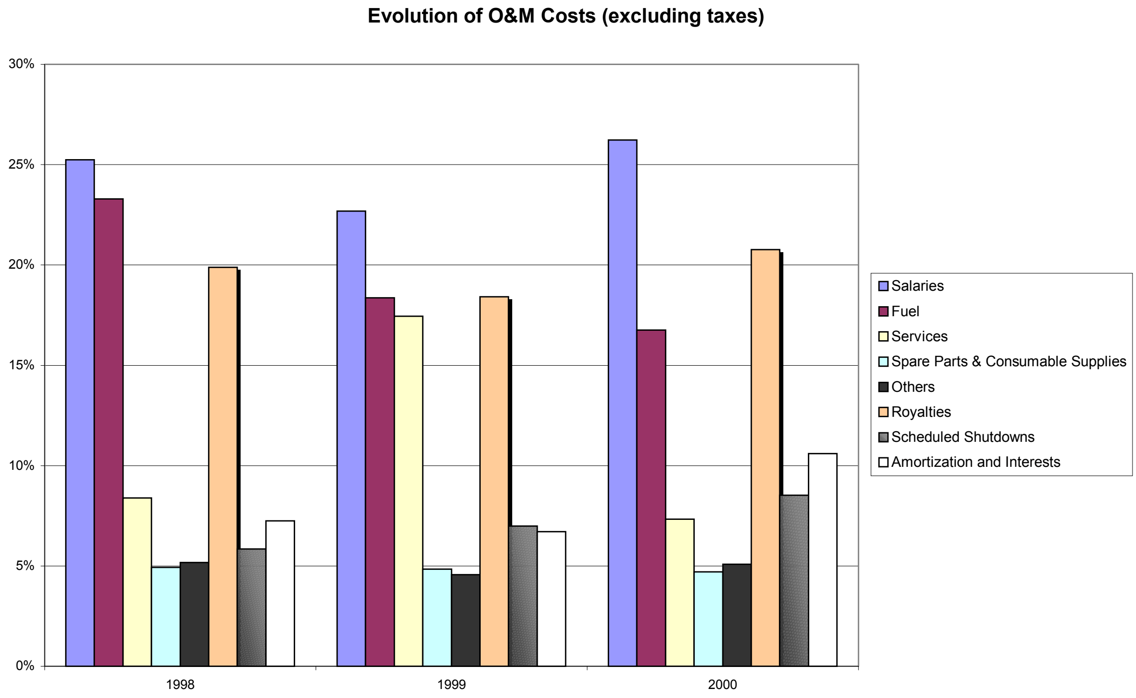
In the following chart the operation and maintenance costs are presented, discriminating among the main items for years 1998, 1999 and 2000.

Evolution of O&M Costs
(in $ x 1000000)
Period 1998 - 2000

Title/Year	1998	1999	2000
Salaries	41,5	40,3	41,8
Fuel	38,3	32,6	26,7
Services	13,8	31,0	11,7
Spare Parts and Consumable Supplies	8,1	8,6	7,5
Others	8,5	8,1	8,1
Royalties	32,7	32,7	33,1
Scheduled Shutdowns	9,6	12,4	13,6
Amortization and Interests	11,9	11,9	16,9
Total	**164,4**	**177,6**	**159,4**

Note: taxes are not included

The following figure shows the evolution of main titles in O&M costs (in %).



Additional salary expenses are shown during 2000, due to a great demand in overtime work because of Atucha I outage and scheduled shutdowns.

Fuel expenses decrease progressively as a result of renegotiations of supply contracts and savings made in Atucha I out of slightly enriched U consumption instead of Natural U.

11.5 Investments on Improving and Upgrading of Nuclear Power Plants Components and Systems

11.5.1 Atucha I

The investments for improving and upgrading the electromechanical components and systems take place continually during operation and the planned plant outages.

The objectives to improve and update are the following:

Internal Components of the Reactor:

- ✓ Safety Study of the Pressure Vessel
- ✓ Manufacturing of the Control Rods Guide Tubes
- ✓ Manufacturing of the Neutron flux Sensors of the Guide Tubes
- ✓ Neutron flux measurements devices
- ✓ Auxiliary devices

Second Heat Sink

- ✓ Basic and Detailed Engineering
- ✓ Electromechanical Components
- ✓ Instrumentation and Control
- ✓ Civil Works
- ✓ Assembling

Other improvements

- ✓ Design modification of the main primary pump seals
- ✓ Instrumentation and control modernization.
- ✓ Design changes on boron shutdown system
- ✓ Modification of pool spent fuel capacity
- ✓ Radiological Protection Equipment
- ✓ Moderator Exchanger Repair Equipment
- ✓ Physical Safety Equipment
- ✓ Building modifications
- ✓ Equipments, Instruments and Tools
- ✓ Hardware and Software
- ✓ Transportation

11.5.2 Embalse

Primary System and Moderator

- ✓ Garter springs repositioning
- ✓ Pipes special Inspections
- ✓ Replacement of the Separator plates of the Steam Generators
- ✓ Replacement of Neutron flux Detectors
- ✓ Support installation on moderator heat exchangers

Electric Systems and Instrumentation and Control

- ✓ Modernization of Protections
- ✓ Renewal of Batteries
- ✓ Replacement of Electric Engines

Waste Storage

- ✓ Improvements of low and intermediate activity waste management

Updating

- ✓ Software and Codes
- ✓ Computational Equipments

11.6 Personnel Qualification

Both Power Plants continued fulfilling the training programs of the personnel at the plants, through courses, simulator practices, operative incidents analysis, exchange of experience with personnel from other plants (IAEA Program), practices at maintenance scale models and inspections in high exposure areas, etc.

The Regulatory Body wrote Standard AR 0.11.3 about “Training of Relevant Installations Personnel”, Revision that came into force in 2000. The said Standard establishes a series of requirements about the training of licensing personnel that renews the specific authorization (see 7.3).

11.7 Deregulation and Privatization of Electric Market

The Argentine National Government launched in 1989 a process of transformation in the role of the State. The main objective in this process was to transfer to the private sector those state companies dedicated to produce goods and services. Based on considering as possible and convenient for the society the private sector competitiveness, it was implemented through a deregulation and privatisation program for transportation, distribution, and electricity generation. In particular, the restructuring of the nuclear sector started in 1994, two years later than the electrical sector.

A period of transformations has begun for the new organizations after Atomic Energy National Commission splitting (see 6.1 NNSR 1998). The resulting new Atomic Energy National Commission has adapted its activity programs to new responsibilities according to the Nuclear Act, and to the assigned annual budgets.

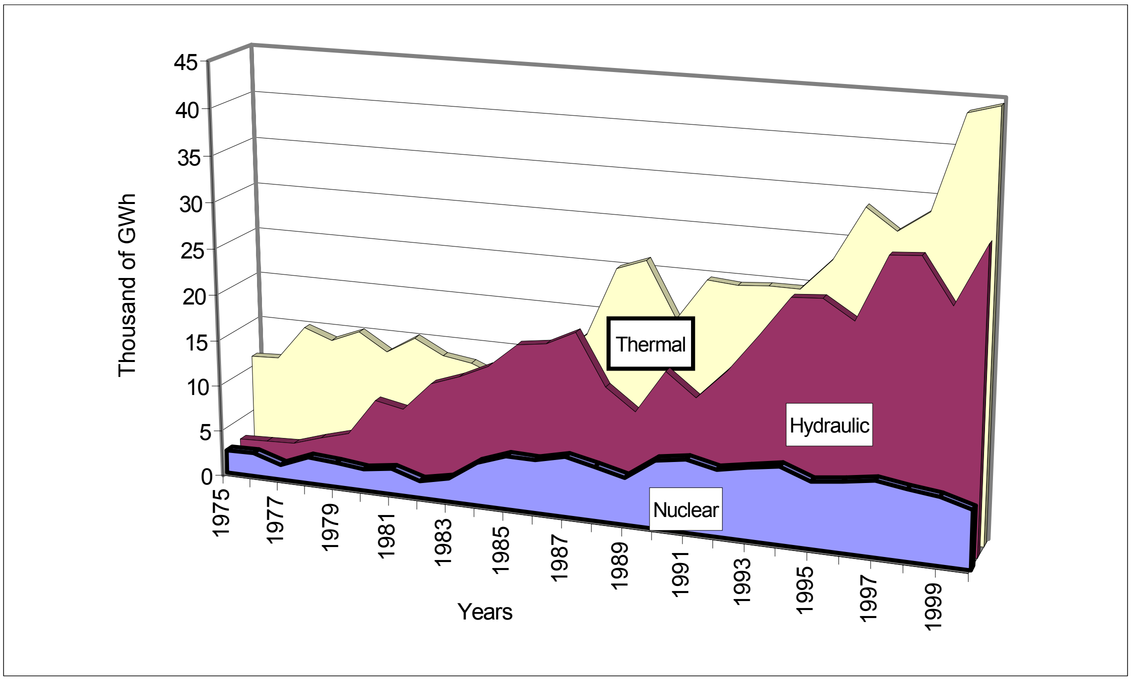
The Regulatory Body has improved its capability to supervise the nuclear power plants, reinforcing its departments of nuclear safety analysis and inspectorate. Besides, an important change has been made by Nucleoeléctrica Argentina S.A. to compete in an open electric market. Nucleoeléctrica Argentina S.A. has its own organization to cope with the requirements and necessities of any self-sustained electric generation company. This transformation has also a key reference: the progressive drop of the kWh price in the generation spot market and the prospective for future prizing.

At present, the main activities of Nucleoeléctrica Argentina S.A. headquarters are:

- ✓ Analysis of future generation.
- ✓ Planning of annual generation and of optimum schedule of nuclear power plants shutdowns according to maintenance programs.
- ✓ Assessment of economic resources produced by selling power and electricity.
- ✓ Daily spot price surveillance for optimal dispatch.
- ✓ Budget preparation and follow up of its fulfillment.

- ✓ Technical and manpower service contacts and purchase orders for goods and spare pacts. Contracts and purchases are made by national and foreign suppliers.
- ✓ Technical support assistance such as purchase specifications, NPP's design, modifications, answers to *ARN*, etc.

Nuclear Power Plants participation in total generation per year has been:



These figures show that in average, the production after 1994 is higher than before. Even so, the resources obtained from the sale of energy and power did not follow the production figures, reflecting the progressive kWh price drop after electrical market privatization. Before 1994 *NASA* received more than 32 mils per kWh. In 1998 the average has been less than 24 mils per kWh.

The effect of this open market on the nuclear power plants has been a strong motivation to improve management efficiency and to decrease production costs. To achieve these objectives, both nuclear power plants have implemented changes to get a better use of human resources in operation and maintenance.

Each nuclear power plant has a Costs Department for planning future activities. The action of these Departments is especially important when the technical sectors are scheduling outages for maintenance. The target is the reduction of outage times, optimizing external services, particularly foreign services, from the main components suppliers (turbine, generator, pumps, etc.) without causing any impact on the safety of the plants.

Another aspect to be analyzed before taking decisions is the main plant design modifications. There are two categories of modifications: those tending to improve the safety of the installation or to facilitate maintenance and those for the nuclear safety upgrading that are required by the Regulatory Body. For the first, a sound technical and economic analysis is made about the cost and benefits, before the approval. For the second, the utilization of probabilistic safety analysis (PSA) results helps in taking decisions about the sequence of modifications. It tends to optimize investments in nuclear safety.

The PSA results are accepted by the regulator, coming to an agreement about the priorities. The adoption of ICRP recommendations -regulatory requirement- led the nuclear power plants to make more efforts on the ALARA program and to introduce the radiological dose costs in maintenance and inspections planning. To reduce dose costs, the use of mock up for training has been intensified, for these jobs to be made in high radiation fields.

But the most important contribution to reduce operating costs is the use of slightly enriched uranium fuel to get higher burn-up firstly implemented in Atucha I power plant. This program started in 1994 and consists in replacing progressively natural Uranium fuel by 0,85% U235 fuel. The program has been very successful increasing up the burn-up from 5900 MWd/t to 11300 MWd/t. The higher burn-up reduces daily refueling from 1,31 to 0,70 fuel element per full power day, reducing also operation and maintenance cost of refueling system. Besides, this refueling reduction extends in time the spent fuel pool capacity.

The main target for the utilities is to achieve competitive generation costs through the following:

- technical changes
  - ✓ The use of slightly enriched Uranium to improve the Atucha nuclear power plants low burn-up.
  - ✓ Reduction time in scheduled outages of nuclear power plants.
  - ✓ Modifications and upgrading in nuclear power plants using probabilistic safety analysis results.
  - ✓ Company Personnel restructuring.
- Cultural changes
  - ✓ Awareness of a commercial competitive activity.
  - ✓ ALARA criteria strengthening.
  - ✓ Benchmarking for specific jobs training.
  - ✓ More intensive discussions with regulators.

Argentine electric industry privatization carried out competitiveness in generation. Tariff dropped down in six years. Anyway Nucleoeléctrica Argentina S.A. has been able to compete in the new electric open market. It was possible because:

- ✓ The operating costs of nuclear power plants were lower than average generation price before privatization of the electric market, and
- ✓ *NASA* implemented organizational and management improvements increasing generation with lower costs. Improvements are not finished because competitiveness has no end.

ARTICLE 12
HUMAN FACTORS

The human factors were discussed in detail in the first report. Methods to prevent, detect and correct the occurrence of events related with human factors are being used in the Operating Organization. The Operating Organization has established a program to evaluate the incidence of human factors in Safety Performance of the Nuclear Power Plants. The program mainly comprises:

1. Evaluation of low safety significant events, near misses and operational events to find out human factors related causes in a systematic way. Evaluation of low level events and near misses are used as a learning method to prevent occurrence of safety significant events.
2. Lessons learned from such events are used as feed back to prevent recurrence. To reach this objective, plant personnel receive specific training as a consequence of events.
3. Identification of precursors to operational events. Some human factors related to low safety significant events and “near misses” can be correlated with organizational deficiencies, therefore enable low safety significant events and “near misses” to be used as leading indicators in anticipating and preventing declining performance.
4. Detection of Organizational (human-related) deficiencies shows how Safety must be managed to help avoiding mistakes and preventing incidents.

To evaluate incidence of human factors in low level events, near misses and operational events, the most commonly used methods were adjusted to different situations. The systematisation of the evaluation process involves the use of Internationally applicable methodologies to evaluate human performance in Atucha I and Embalse nuclear power plants.

Combination of Human Performance Enhancement System, Human Performance Investigation Program, and Assessment of Safety Significant Events Techniques methodologies are still considered enough to detect both human factors and organizational deficiencies as “root causes” and “contributing causes” of analysed events.

Using those methodologies and their associated techniques 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, containing plant abnormalities

- ✓ Safety indicators like collective dose, surveillance deviations
- ✓ Use and procedures and instructions
- ✓ Organizational and individual Safety Culture

For Regulatory Body technical working groups, events screening and analysis is an useful tool when it is necessary to request:

- ✓ Training or re-training of plants personnel
- ✓ Updating of procedures
- ✓ Generation of changes in managerial attitude looking for Safety Culture
- ✓ Investigation of detected precursors of significant events

Lessons learned and corrective actions are followed through inspections and regulatory audits. However, considering the experience gathered during the first stage of Atucha I probabilistic safety analysis (PSA) and the periodic training of Embalse nuclear power plant at Gentilly-II simulator in Canada, as well as the human reliability analyses carried out since 1998 for the Embalse nuclear power plant level 1 PSA and for Atucha I PSA 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.

ARTICLE 13
QUALITY ASSURANCE

13.1 Introduction

As it was stated in the first report, the application of adequate quality assurance programs in the stages of design, construction, operation, and decommissioning of nuclear installations is a regulatory requirement in Argentina. In the case of Nuclear Power Plants, standard AR 3.6.1. (1) establishes the requisites that must be fulfilled. Moreover, Standard AR 3.7.1 determines when the Responsible Organization must present the program and Quality Assurance Manual.

Additionally, the operation licences of Atucha I and Embalse nuclear power plants establish that both plants must have quality assurance programs. Particularly, and as an additional quality assurance requisite, the Embalse Nuclear Power Plant requires special procedures to modify the control room computer system software. Atucha II construction licence includes, among other requirements, a quality assurance program within that stage. In all cases quality assurance programs and manuals have, among other documents, mandatory force for the installation.

The Regulatory Body supervises the implementation of the Quality Assurance Programs of the Responsible Organization through audits or the results obtained from special inspections about activities such as documentation control, disagreements control and tasks execution.

13.2 Licence Official Quality Assurance Program

Argentine Nuclear Power Plants in operation or in construction have quality assurance programs that are documented, implemented, revised and evaluated by the Plant Management. The frame of these specific programs is the General Quality Assurance Program of the Responsible Organization.

Regarding the Quality Assurance documentation, due to the fact that the General Manual was developed after specific Programs, the organization units had to adapt its specific manuals and procedures in accordance with the new requirements, since the General Manual was approved.

Therefore, Atucha I made the 2nd Revision of the Quality Assurance Manual for the Operation, which is being evaluated by the Regulatory Body at present. Regarding the previous version, this revision considers what has been established in the Responsible Organization Quality Assurance Manual and fulfils the requirements of Regulatory Standard AR 3.6.1, IAEA Practice Code 50-C-Q and all IAEA applicable Safety Guides.

The same happens with the new versions of the Quality Assurance Manual for the construction and the Operation of both Atucha I and Embalse nuclear power plants, respectively. Table 13.1 shows the state of the Quality Assurance General program updated on January 2001.

Table 13.1

ORGANIZATION UNIT	DOCUMENT	REVISION	PROCEDURE NUMBER
NASA	General Manual	Revision 0 Updated	General Procedure 10
CNA I	Quality Assurance Manual for the Operation	Revision 2(*)	200
CNE	Quality Assurance Manual for the Operation	Revision 4 Updated	460
CNA II	Quality Assurance Manual for the Construction	Revision 2 Updated	60
Engineering and Support Services	Services Dept. Quality Assurance Manual	Revision 4 Updated	150
ULE Project (**)	Quality Assurance Manual for the ULE Project	Revision 3 Updated	8

(*) In process of evaluation to be approved.
(**) Fuel elements with slightly enriched uranium

The evaluation of the Program implementation at the Plants is the Quality Assurance Divisions´ responsability, reporting the results of both units to the Manager. On the other hand, the Quality Assurance Department is the responsible for the evaluation of the Installations General and Specific Programs implementation. Periodically, the mentioned Divisions write down reports including the results of the audits and surveillance, which are sent to the Installation Manager and the Responsible Organization General Manager.

Figures. 13.1 y 13.2 show the organization charts updated and approved by Atucha I and Embalse Managements, respectively. Atucha I organization chart has remained without any change since 1998, but Embalse organization has suffered significant modifications. This does not affect the Quality Assurance Division that maintains its direct dependence on the Plant Manager.

FIGURE 13.1 ATUCHA I NPP ORGANIZATION CHART

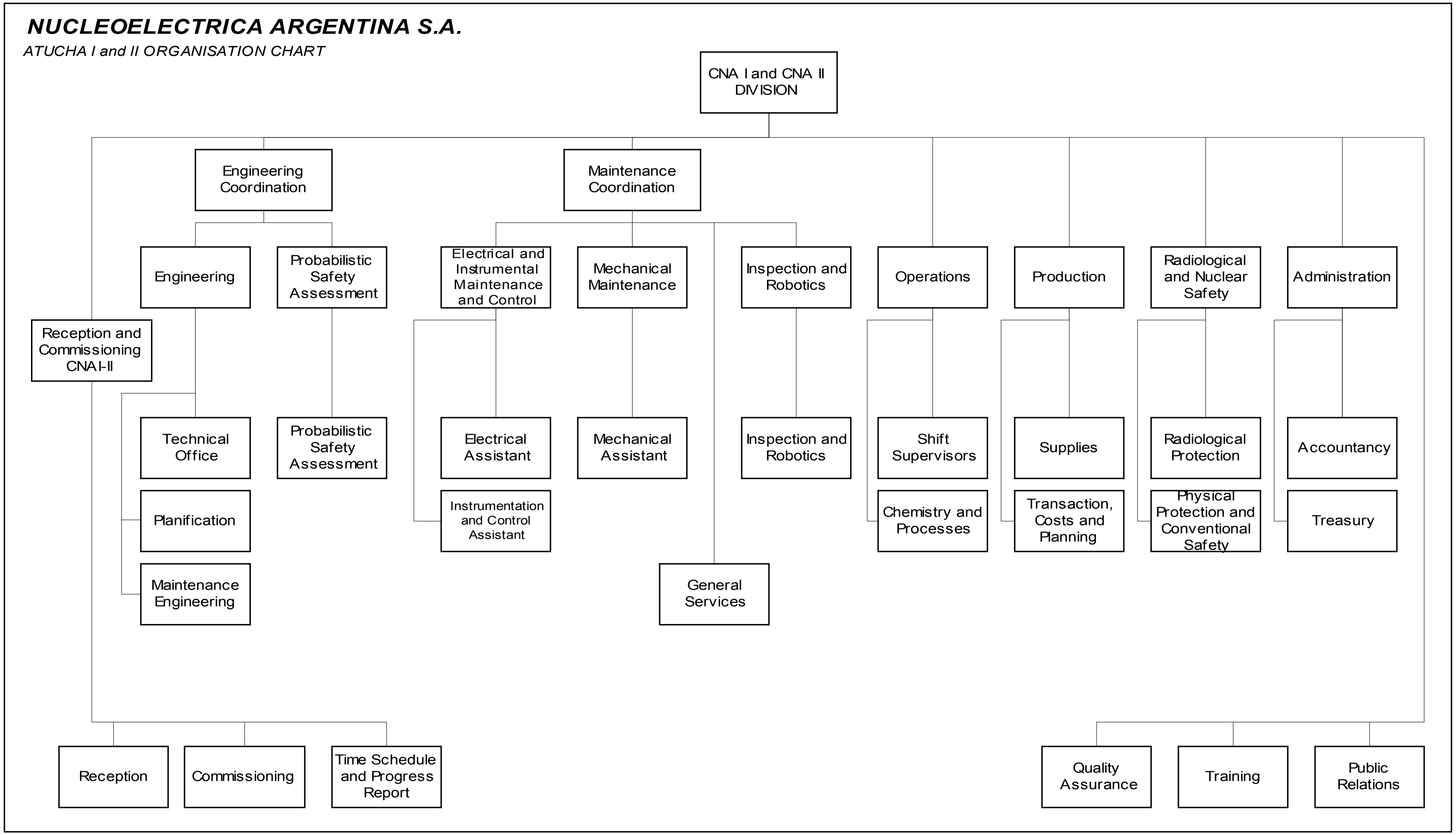
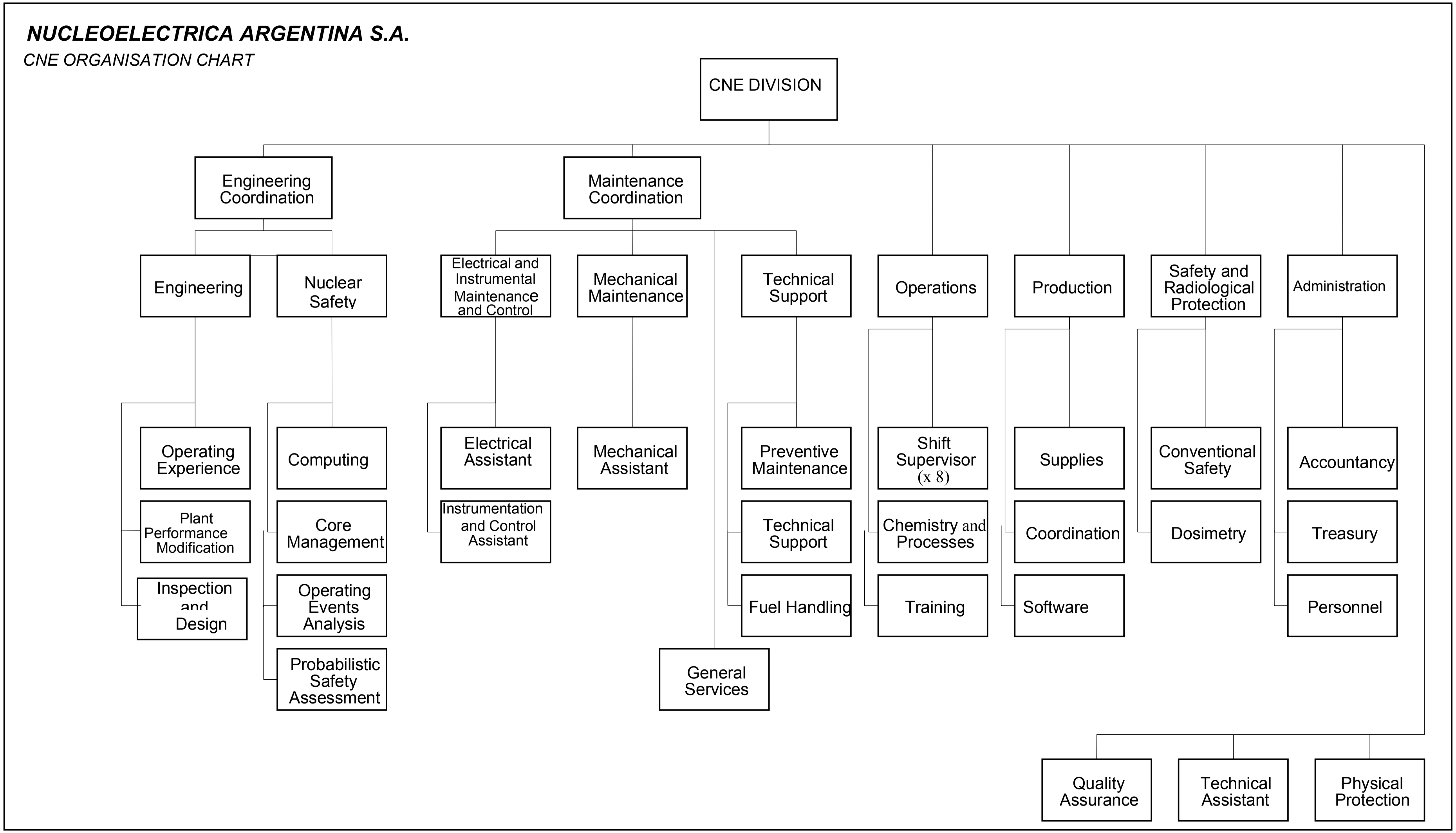


FIGURE 13.2 EMBALSE NPP ORGANIZATION CHART



ARTICLE 14
ASSESSMENT AND VERIFICATION OF SAFETY

The safety assessment of both nuclear power plants, inspections and audits were detailed in the First National Nuclear Safety Report by the Regulatory Body and the Responsible Organization. In this report the main safety assessment efforts were explained in detail within Article 6. However, from the first review meeting significant evaluations were performed in the area of ageing and regulatory plant safety indicators.

14.1 Ageing

Each argentine nuclear power plant in operation has implemented an own Ageing Management Program required by the Regulatory Body with the objective of maintaining the safety of the nuclear power plants lifetime by optimising the inspection and maintenance programs, replacing parts and ageing monitoring, prevention and mitigation. The Ageing Management Program is useful in preventing and detecting systematically any degradation that involves equipment, systems and components by affecting the design safety margins. The Ageing Management Program 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.

As a result of the Ageing Management Program application, the following activities were performed:

Atucha I Nuclear Power Plant

The selected components to be followed up into the Ageing Management Program are valves, pipelines, pumps and heat exchangers belonging to the primary system, moderator system, feed and bleed system, shutdown systems, emergency core cooling system, reactor system, I&C and electric systems.

Ageing of specific components like reactor pressure vessel, fuel channels, fuel channel thermal isolation, moderator tank, tube guide control rods and control rods are described and analysed in Article 6.

Embalse Nuclear Power Plant

The selected components to be followed up within the Ageing Management Program are valves, pipelines, pumps and heat exchangers belonging to the primary system, pressure and inventory control system, shutdown systems, reactor system, emergency core cooling system, Emergency water supply, containment building (concrete), feedwater system and dousing system.

Pressure tubes: Pressure Tubes suffer a progressive degradation due mainly to the following mechanisms: hydrogen uptake, blister formation, rolled joint cracked and bend/deformation. The Regulatory Body required to constrain plant operation to fulfil the following items:

- ✓ To demonstrate that there are not pressure tubes containing hydrogen / deuterium over the blister formation threshold.
- ✓ To present a new garter spring repositioning program for the pressure tubes that remain without being repositioned.

The above mentioned requirement was fulfilled by performing scrapping to ten selected pressure tubes. The scrapping samples results indicated that no one pressure tube containing hydrogen / deuterium over blister formation threshold would be present until October 2004 (see 6.2.1).

Up to now 201 pressure tubes (53% of the whole number) were inspected, 195 pressure tubes have been repositioned and 179 pressure tubes (47% of the whole number) remain to be inspected. Up to the year 2000 pressure/calandria tubes contact was observed in 25% of the tubes. Inspections were performed according to CAN/CSA – N 285.4 – 94 Canadian Standard.

Feeders: The inspection was performed according to what it had been established by the In-service Inspection program during scheduled outages (see 6.2.2).

Mercury wetted relays: An increase in the relay failures number was detected and it was decided to analyse the causes indicating that mercury has been suffering a degradation process (ageing) that provokes the sticking of the mercury with the contacts. It means that contacts will remain closed. At that time, all Embalse nuclear power plant safety systems had this kind of relays. To reduce the relay failures rates, the Responsible Organization replaced the mercury-wetted relays by mercury wetted relays doped with tin.

14.2 Regulatory Plant Safety Indicators

Additionally to the plant safety indicators program carried out by the Responsible Organization, the Regulatory Body considers the use of numerical indicators as an important tool to evaluate the safety performance of nuclear powers plants. For this reason, in 1997 the Regulatory Body established a program in order to define and implement a new system of Safety Performance Indicators for the two operating nuclear power plants, Atucha I and Embalse .

In 1998 the first set of indicators was defined and a pilot implementation program was initiated. As a result of the pilot program, several improvements have been carried out.

The objective of the indicators program is to incorporate a set of safety performance indicators to be used as a new regulatory tool to provide an additional view of the operational performance of the nuclear power plants, improving the ability to detect degradation on safety related areas.

To be useful indicators have to be predictive and sensitive. Considering the effects that degradations at organizational and programmatic level produce, indirect indicators have been included in order to evaluate those aspects. Taking into account the objective of the program, the system of performance indicators is constrained to safety related areas.

Definition of a set of indicators:

Before defining the set of performance indicators, it was necessary to define a framework to be sure all areas of interest have been included. Besides, it is useful to search orderly the indicators.

There are several methodologies to define that framework, some of them, used by other organizations were analysed. Using these references, a framework was defined. It has been divided into six main areas: Operation Stability, Radiation Protection, Surveillance (Maintenance and Testing), Organization, Abnormal Operation and Risk.

The next step was to identify relevant aspects and to define indicators in each area. The most important limitation to define and include an indicator was to quantify it. The feasibility of getting information was also considered. For this reason, the first set of indicators was discussed and agreed with the plant personnel. The set of indicators is elaborated quarterly.

ARTICLE 15
RADIOLOGICAL PROTECTION

There has not been any change in the general criteria and standards used in Radiological Protection. The environmental impact in Argentina has not been modified due to the operation of the Nuclear Power Plants, nor have the parameters that have been used been significantly modified.

15.1 Atucha I Nuclear Power Plant

15.1.1 Radioactive release into the environment

The limits of the environment release of Atucha I nuclear power plant have not been modified.

Table 15.1 –Authorised gaseous discharge limits for Atucha I

NUCLIDE	K_i (TBq)
Sr-89	2×10^{0}
Cs-134	5×10^{-2}
H-3	1×10^{4}
Kr-85m	6×10^{3}
Kr-88	5×10^{2}
Ba-140	5×10^{0}
Ru-106	3×10^{-1}
Sb-124	1×10^{0}
Xe-133	3×10^{4}
Ar-41	7×10^{2}
Co-60	1×10^{-1}
Cs-137	3×10^{-2}
I-131	4×10^{-2}
Kr-87	7×10^{2}
Transuranides	2×10^{-3}
Ru-103	5×10^{0}
Sb-122	1×10^{1}
Sr-90	4×10^{-2}
Xe-135	4×10^{3}

Table 15.2 – Authorised liquid discharge limits for Atucha I

NUCLIDE	K_i (TBq)
Ba-140	4×10^{2}
Co-60	1×10^{1}
Cs-134	6×10^{-1}
Fe-59	9×10^{1}
I-131	2×10^{1}
Mn-54	6×10^{1}
Sb-125	1×10^{2}
Ru-106	9×10^{1}
Sb-124	3×10^{2}
Sr-90	1×10^{1}
Zr-95	6×10^{1}
Ce-144	6×10^{1}
Ag-110m	8×10^{1}
Cr-51	2×10^{3}
Cs-137	7×10^{-1}
H-3	1×10^{5}
Transuranides	5×10^{0}
Ni-65	2×10^{4}
Ru-103	7×10^{2}
Sb-122	4×10^{2}
Sr-89	8×10^{1}
Zn-65	6×10^{0}
Co-58	7×10^{1}

The gaseous radioactive releases to the environment due to Atucha I nuclear power plant operation since its start-up may be observed in Table 15.3, discriminating those corresponding to I-131, H-3, aerosols and noble gases; it also includes an estimation of C-14 discharge.

Table 15.3 – Activity released from Atucha I to the environment as gaseous discharges

YEAR	I-131 (TBq)	Tritium (TBq)	Aerosols (TBq)	Noble Gases (TBq)	C-14 Estimated values (TBq)
1974	3.0×10^{-4}	8.0×10^{0}	2.3×10^{-6}	6.7×10^{1}	4.0×10^{-1}
1975	4.6×10^{-5}	3.8×10^{1}	5.4×10^{-6}	9.3×10^{0}	4.0×10^{-1}
1976	3.6×10^{-4}	2.2×10^{2}	1.1×10^{-5}	1.6×10^{2}	4.0×10^{-1}
1977	4.3×10^{-5}	2.2×10^{2}	5.3×10^{-6}	7.0×10^{1}	3.0×10^{-1}
1978	1.8×10^{-3}	2.2×10^{2}	2.0×10^{-5}	3.1×10^{2}	5.0×10^{-1}
1979	2.7×10^{-3}	2.3×10^{2}	2.4×10^{-5}	2.5×10^{2}	4.5×10^{-1}
1980	2.0×10^{-4}	2.4×10^{2}	1.6×10^{-5}	2.5×10^{2}	4.2×10^{-1}
1981	4.2×10^{-4}	2.1×10^{2}	1.4×10^{-5}	4.6×10^{1}	4.8×10^{-1}
1982	1.9×10^{-5}	3.0×10^{2}	7.4×10^{-6}	1.9×10^{1}	4.5×10^{-1}
1983	1.4×10^{-4}	6.3×10^{2}	8.1×10^{-6}	4.7×10^{1}	4.3×10^{-1}
1984	9.2×10^{-6}	2.0×10^{2}	4.4×10^{-6}	9.0×10^{-1}	5.0×10^{-1}
1985	5.9×10^{-4}	2.5×10^{2}	2.2×10^{-5}	5.5×10^{0}	3.7×10^{-1}
1986	5.9×10^{-4}	3.2×10^{2}	4.4×10^{-6}	6.2×10^{0}	3.8×10^{-1}
1987	6.5×10^{-5}	4.6×10^{2}	1.4×10^{-5}	1.4×10^{0}	2.7×10^{-1}
1988	2.3×10^{-4}	8.1×10^{2}	2.3×10^{-6}	3.5×10^{0}	1.5×10^{-1}
1989	1.3×10^{-6}	7.0×10^{2}	7.6×10^{-7}	5.6×10^{-1}	0
1990	7.8×10^{-5}	6.2×10^{2}	1.1×10^{-6}	8.9×10^{1}	3.3×10^{-1}
1991	1.3×10^{-3}	2.3×10^{2}	1.5×10^{-5}	1.1×10^{1}	5.2×10^{-1}
1992	8.9×10^{-6}	4.1×10^{2}	1.5×10^{-5}	3.0×10^{0}	4.5×10^{-1}
1993	4.9×10^{-4}	2.6×10^{3}	1.8×10^{-4}	1.1×10^{2}	4.9×10^{-1}
1994	4.4×10^{-4}	1.4×10^{3}	4.9×10^{-5}	2.4×10^{2}	5.2×10^{-1}
1995	3.5×10^{-4}	5.3×10^{2}	1.3×10^{-5}	3.6×10^{2}	5.5×10^{-1}
1996	4.1×10^{-5}	1.1×10^{3}	3.8×10^{-5}	3.2×10^{2}	4.3×10^{-1}
1997	5.3×10^{-4}	1.3×10^{3}	6.0×10^{-6}	9.6×10^{2}	5.6×10^{-1}
1998	7.9×10^{-6}	3.9×10^{2}	8.0×10^{-6}	1.3×10^{2}	4.9×10^{-1}
1999	1.9×10^{-6}	8.2×10^{2}	2.6×10^{-6}	2.9×10^{1}	2.9×10^{-1}
2000	6.5×10^{-5}	1.2×10^{3}	5.5×10^{-6}	7.4×10^{1}	3.4×10^{-1}
AVERAGE	4.0×10^{-4}	5.8×10^{2}	1.8×10^{-5}	1.3×10^{2}	4.0×10^{-1}

The liquid radioactive releases to the environment by Atucha I nuclear power plant since its start-up until December 2000 are presented in Table 15.4, discriminating between liquid discharges of H-3 and gamma emitters.

The 89% of the total average discharge from Atucha I nuclear power plant to the environment corresponded to tritium. Comparing these discharges with the respective annual authorised discharge limits, it is observed that they were less than 6% of such limits.

Table 15.4 – Activity released from Atucha I to the environment as liquid discharges

YEAR	TRITIUM (TBq)	GAMMA EMITTERS (TBq)
1974	3.3×10^{0}	5.2×10^{-2}
1975	3.1×10^{1}	1.5×10^{-1}
1976	8.1×10^{1}	2.2×10^{-1}
1977	2.2×10^{2}	1.1×10^{-1}
1978	2.3×10^{2}	7.8×10^{-2}
1979	2.6×10^{2}	1.2×10^{-1}
1980	2.9×10^{2}	8.2×10^{-2}
1981	4.1×10^{2}	8.1×10^{-2}
1982	3.1×10^{2}	5.1×10^{-2}
1983	2.4×10^{2}	3.7×10^{-2}
1984	4.1×10^{2}	5.1×10^{-2}
1985	3.2×10^{2}	5.1×10^{-2}
1986	2.8×10^{2}	4.2×10^{-2}
1987	3.6×10^{2}	1.0×10^{-1}
1988	5.9×10^{2}	9.6×10^{-2}
1989	3.0×10^{2}	5.9×10^{-2}
1990	5.3×10^{2}	1.3×10^{-1}
1991	5.5×10^{2}	9.3×10^{-2}
1992	7.7×10^{2}	9.3×10^{-2}
1993	9.2×10^{2}	6.0×10^{-2}
1994	2.2×10^{3}	6.6×10^{-1}
1995	5.0×10^{2}	3.3×10^{-1}
1996	5.5×10^{2}	6.8×10^{-1}
1997	1.2×10^{3}	2.3×10^{-1}
1998	6.9×10^{2}	1.3×10^{-1}
1999	$8. \times 10^{2}$	3.5×10^{-1}
2000	8.4×10^{2}	3.3×10^{-1}
AVERAGE	5.1×10^{2}	1.7×10^{-1}

15.1.2 Public exposure

The annual average dose to the critical group due to Atucha I nuclear power plant operation, during the period 1974-2000, was lower than 2% of the established individual dose constraint. Gaseous discharges were the main contributor.

The annual dose values to the critical group for the period 1974-2000 are shown in Table 15.5, discriminated according to the discharge type.

TABLE 15.5 –CRITICAL GROUP INDIVIDUAL DOSE FOR ATUCHA I

YEAR	DUE TO GASEOUS DISCHARGES (MSV)	DUE TO LIQUID DISCHARGES (MSV)	TOTAL DOSE (MSV)
1974	6.0×10^{-4}	5.8×10^{-4}	1.2×10^{-3}
1975	2.2×10^{-4}	1.6×10^{-3}	1.8×10^{-3}
1976	2.1×10^{-3}	2.5×10^{-3}	4.6×10^{-3}
1977	1.4×10^{-3}	1.5×10^{-3}	2.9×10^{-3}
1978	3.4×10^{-3}	1.1×10^{-3}	4.5×10^{-3}
1979	3.1×10^{-3}	1.6×10^{-3}	4.7×10^{-3}
1980	2.8×10^{-3}	1.2×10^{-3}	4.0×10^{-3}
1981	1.3×10^{-3}	1.3×10^{-3}	2.6×10^{-3}
1982	1.4×10^{-3}	8.7×10^{-4}	2.3×10^{-3}
1983	3.1×10^{-3}	6.4×10^{-4}	3.7×10^{-3}
1984	9.0×10^{-4}	9.6×10^{-4}	1.9×10^{-3}
1985	1.2×10^{-3}	8.9×10^{-4}	2.1×10^{-3}
1986	1.5×10^{-3}	7.0×10^{-4}	2.2×10^{-3}
1987	2.0×10^{-3}	8.8×10^{-4}	2.9×10^{-3}
1988	3.6×10^{-3}	9.5×10^{-4}	4.5×10^{-3}
1989	3.1×10^{-3}	5.5×10^{-4}	3.7×10^{-3}
1990	3.1×10^{-3}	5.3×10^{-4}	3.6×10^{-3}
1991	2.5×10^{-3}	4.4×10^{-4}	2.9×10^{-3}
1992	1.7×10^{-3}	5.1×10^{-4}	2.2×10^{-3}
1993	1.0×10^{-2}	2.8×10^{-4}	1.0×10^{-2}
1994	7.1×10^{-3}	3.8×10^{-4}	7.5×10^{-3}
1995	6.2×10^{-3}	2.0×10^{-4}	6.4×10^{-3}
1996	8.2×10^{-3}	3.6×10^{-4}	8.6×10^{-3}
1997	1.1×10^{-2}	3.6×10^{-4}	1.1×10^{-2}
1998	3.7×10^{-3}	2.3×10^{-4}	3.9×10^{-3}
1999	3.7×10^{-3}	4.0×10^{-4}	4.1×10^{-3}
2000	4.5×10^{-3}	4.1×10^{-4}	4.9×10^{-3}
AVERAGE	3.5×10^{-3}	8.1×10^{-4}	4.3×10^{-3}

The effective collective dose normalised per unit of electric energy generated $(GW_{(e)}\ y)^{-1}$, is presented in Table 15.6, calculated with population data up to a radius of 2000 km from the nuclear power plant.

Table 15.6 – Regional normalised collective effective dose for Atucha I

YEAR	Due to gaseous discharges [man Sv $(GW_{(e)}\ y)^{-1}$]	Due to liquid discharges [man Sv $(GW_{(e)}\ y)^{-1}$]	Total Dose [man Sv $(GW_{(e)}\ y)^{-1}$]
1974	1.6×10^{-1}	1.7×10^{-1}	3.3×10^{-1}
1975	9.3×10^{-4}	2.7×10^{-1}	2.7×10^{-1}
1976	2.5×10^{-1}	5.1×10^{-1}	7.6×10^{-1}
1977	2.6×10^{-1}	1.3×10^{0}	1.6×10^{0}
1978	3.2×10^{-1}	7.7×10^{-1}	1.1×10^{0}
1979	3.2×10^{-1}	9.8×10^{-1}	1.3×10^{0}
1980	3.2×10^{-1}	1.1×10^{0}	1.4×10^{0}
1981	1.3×10^{-1}	1.4×10^{0}	1.5×10^{0}
1982	2.4×10^{-1}	1.5×10^{0}	1.7×10^{0}
1983	3.7×10^{-1}	8.9×10^{-1}	1.3×10^{0}
1984	1.5×10^{-1}	2.1×10^{0}	2.2×10^{0}
1985	2.2×10^{-1}	2.2×10^{0}	2.4×10^{0}
1986	1.7×10^{-1}	1.0×10^{0}	1.2×10^{0}
1987	3.6×10^{-1}	2.4×10^{0}	2.8×10^{0}
1988	1.3×10^{0}	2.7×10^{0}	4.0×10^{0}
1989	(*)	(*)	(*)
1990	5.0×10^{-1}	1.1×10^{0}	1.6×10^{0}
1991	1.1×10^{-1}	7.1×10^{-1}	8.2×10^{-1}
1992	2.3×10^{-1}	1.2×10^{0}	1.4×10^{0}
1993	1.3×10^{0}	1.0×10^{0}	2.3×10^{0}
1994	7.5×10^{-1}	1.6×10^{0}	2.3×10^{0}
1995	3.1×10^{-1}	3.1×10^{-1}	6.2×10^{-1}
1996	7.2×10^{-1}	4.4×10^{-1}	1.2×10^{0}
1997	7.6×10^{-1}	6.9×10^{-1}	1.4×10^{0}
1998	2.4×10^{-1}	5.5×10^{-1}	8.0×10^{-1}
1999	7.1×10^{-1}	1.2×10^{0}	1.9×10^{0}
2000	8.8×10^{-1}	1.1×10^{0}	2.0×10^{0}
AVERAGE	4.3×10^{-1}	1.1×10^{0}	1.5×10^{0}

(*) Electrical energy was not generated

The average collective effective dose per unit of electric energy generated, calculated with population data up to a radius of 2000 km from the Atucha I nuclear power plant nuclear power plant, for the period 1974-2000, represented about 10% of the collective effective dose constraint per unit of electric energy generated set by the Regulatory Body in 15 [man Sv $(GW_{(e)}\ y)^{-1}$].

Besides, the average collective effective dose per unit of electric energy generated due to radionuclides of global distribution, was 0.6 [man Sv $(GW_{(e)}\ y)^{-1}$] for tritium and 35 [man Sv $(GW_{(e)}\ y)^{-1}$] for C-14 for the period 1974-2000. Those collective effective doses correspond to the incomplete effective dose commitment integrated over the expected duration of the practice (500 years).

The average normalised effective collective dose for the period 1974-2000, due to C-14 releases is higher than the collective effective dose constraint per unit of electric energy generated established in Standard 3.1.2. This is due to the fact that Atucha I nuclear power plant initiated its operation before the above mentioned standard was in force.

15.2 Embalse Nuclear Power Plant

15.2.1 Radioactive release into the environment

The Regulatory Body authorised a set of gaseous and liquid limits, contained in the plant operating license and shown in Tables 15.7 and 15.8 respectively. For critical group doses, these limits were set much lower than 0.3 mSv.

Table 15.7. Authorised gaseous discharge limits for Embalse

NUCLIDE	K_i (TBq)
Ar-41	7.4×10^{3}
Kr-85m	3.7×10^{4}
Kr-87	7.4×10^{3}
Kr-88	3.7×10^{3}
Xe-133	1.9×10^{5}
Xe-135	3.7×10^{4}
H-3	3.7×10^{4}
I-131	2.2×10^{1}
Co-58	3.7×10^{1}
Co-60	3.7×10^{-1}
Sr-89	1.1×10^{2}
Sr-90	3.7×10^{0}
Ru-106	1.5×10^{0}
Cs-134	1.5×10^{0}
Cs-137	3.7×10^{-1}
Ba-140	1.5×10^{2}

Table 15.8. Authorised liquid discharge limits for Embalse

NUCLIDE	K_i (TBq)
H-3	3.7×10^{3}
Cr-51	3.7×10^{2}
Mn-54	7.4×10^{-1}
Fe-59	3.7×10^{1}
Co-60	1.5×10^{-1}
Zn-65	7.4×10^{-2}
Ni-65	7.4×10^{3}
Sr-89	3.7×10^{0}
Sr-90	1.5×10^{-1}
Zr-95	1.9×10^{0}
Ru-103	3.7×10^{0}
Ru-106	1.5×10^{-1}
Ag-110m	1.1×10^{0}
Sb-125	1.1×10^{0}
I-131	1.9×10^{-1}
Cs-134	3.7×10^{-2}
Cs-137	3.7×10^{-2}
Ba-140	1.1×10^{1}
Ce-144	1.9×10^{-1}
Gd-153	3.0×10^{1}

The gaseous radioactive releases by Embalse to the environment, since its initial operation may be seen in Table 15.9, discriminating those corresponding to I-131, H-3, aerosols and noble gases, and including an estimation of C-14 discharges.

The liquid discharges released by Embalse to the environment from the beginning up to December 2000 are presented in Table 15.10, discriminating between liquid discharges of H-3 and gamma emitters.

The 47% of the total average discharge from Embalse to the environment corresponded to tritium, and 53% to noble gases. These discharges were less than 10% of the annual authorised discharge limit.

On February, 2000 started to operate a liquid discharge evaporator system that, as can be seen in table 15.10, allowed to reduce the liquid tritium discharges to the Río Tercero lake. The reason to introduce the evaporator was to avoid lake tritium concentration increases because the lake is formed by a dam on Río Tercero which constitutes a close watershed during some months over the year.

Table 15.9 – Activity released from Embalse to the environment as gaseous discharges

YEAR	I-131 (TBq)	Tritium (TBq)	Aerosols (TBq)	Noble Gases (TBq)	C-14 Estimated values (TBq)
1984	0	7.3×10^{0}	0	4.1×10^{1}	2.8×10^{-1}
1985	1.9×10^{-3}	3.0×10^{1}	2.2×10^{-4}	1.5×10^{3}	3.9×10^{-1}
1986	2.5×10^{-3}	2.7×10^{1}	3.9×10^{-5}	4.2×10^{2}	3.2×10^{-1}
1987	1.9×10^{-6}	3.3×10^{1}	7.8×10^{-2}	3.1×10^{2}	4.7×10^{-1}
1988	3.7×10^{-4}	4.9×10^{1}	8.8×10^{-5}	9.6×10^{1}	4.6×10^{-1}
1989	0	8.6×10^{1}	0	1.3×10^{2}	4.7×10^{-1}
1990	1.4×10^{-3}	7.5×10^{1}	0	6.6×10^{2}	5.5×10^{-1}
1991	1.6×10^{-3}	5.5×10^{1}	1.2×10^{-4}	1.2×10^{3}	5.0×10^{-1}
1992	7.0×10^{-5}	6.9×10^{1}	2.5×10^{-5}	1.5×10^{2}	4.8×10^{-1}
1993	0	1.4×10^{2}	0	4.2×10^{1}	5.3×10^{-1}
1994	2.6×10^{-4}	1.3×10^{2}	3.6×10^{-6}	1.7×10^{1}	5.7×10^{-1}
1995	1.7×10^{-3}	8.3×10^{1}	7.7×10^{-5}	4.4×10^{1}	4.3×10^{-1}
1996	2.7×10^{-4}	6.9×10^{1}	0	1.8×10^{2}	5.4×10^{-1}
1997	0	7.7×10^{1}	0	3.0×10^{1}	5.2×10^{-1}
1998	0	7.2×10^{1}	0	2.1×10^{1}	5.1×10^{-1}
1999	0	7.8×10^{1}	0	1.6×10^{1}	5.8×10^{-1}
2000	0	2.7×10^{2}	5.1×10^{-6}	1.4×10^{1}	3.9×10^{-1}
AVERAGE	5.9×10^{-4}	7.9×10^{1}	4.6×10^{-3}	2.9×10^{2}	4.7×10^{-1}

Note: The value "0" means lower than the minimum detectable level

TABLE 15.10 – ACTIVITY RELEASED FROM EMBALSE TO THE ENVIRONMENT AS LIQUID DISCHARGES

YEAR	TRITIUM (TBQ)	GAMMA EMITTERS (TBQ)
1984	3.5×10^{0}	7.8×10^{-3}
1985	1.6×10^{1}	1.9×10^{-3}
1986	7.9×10^{1}	7.1×10^{-3}
1987	1.6×10^{2}	4.5×10^{-3}
1988	1.7×10^{2}	2.7×10^{-3}
1989	2.2×10^{2}	5.8×10^{-3}
1990	2.2×10^{2}	3.5×10^{-3}
1991	5.2×10^{2}	2.0×10^{-2}
1992	1.6×10^{2}	2.0×10^{-3}
1993	2.0×10^{2}	2.0×10^{-3}
1994	1.4×10^{2}	1.6×10^{-3}
1995	2.3×10^{2}	4.3×10^{-3}
1996	3.2×10^{2}	4.6×10^{-3}
1997	1.6×10^{2}	2.0×10^{-3}
1998	2.2×10^{2}	2.0×10^{-3}
1999	1.4×10^{2}	4.5×10^{-3}
2000	2.0×10^{1}	1.6×10^{-3}
AVERAGE	1.8×10^{2}	4.6×10^{-3}

15.2.2 Public exposure

The annual critical group doses due to Embalse operation during the period 1984-2000, are presented in Table 15.11, discriminated according to discharge type. The annual average critical group doses due to Embalse operation, for the period 1984-2000 resulted lower than 2% of the established individual dose constraint. The liquid discharges were the main contributor.

TABLE 15.11 – CRITICAL GROUP INDIVIDUAL DOSE FOR EMBALSE

YEAR	DUE TO GASEOUS DISCHARGES (MSV)	DUE TO LIQUID DISCHARGES (MSV)	TOTAL DOSE (MSV)
1984	1.6×10^{-5}	3.9×10^{-4}	4.1×10^{-4}
1985	4.8×10^{-4}	7.9×10^{-4}	1.3×10^{-3}
1986	2.4×10^{-4}	2.8×10^{-3}	3.0×10^{-3}
1987	3.7×10^{-4}	9.5×10^{-3}	9.9×10^{-3}
1988	1.7×10^{-4}	5.9×10^{-3}	6.1×10^{-3}
1989	1.8×10^{-4}	6.7×10^{-3}	6.9×10^{-3}
1990	4.5×10^{-4}	6.4×10^{-3}	6.9×10^{-3}
1991	4.1×10^{-4}	1.1×10^{-2}	1.1×10^{-2}
1992	8.4×10^{-5}	4.0×10^{-3}	4.1×10^{-3}
1993	8.0×10^{-5}	5.0×10^{-3}	5.1×10^{-3}
1994	8.1×10^{-5}	3.8×10^{-3}	3.9×10^{-3}
1995	1.1×10^{-4}	5.5×10^{-3}	5.6×10^{-3}
1996	1.0×10^{-4}	6.8×10^{-3}	6.9×10^{-3}
1997	5.4×10^{-5}	4.6×10^{-3}	4.6×10^{-3}
1998	4.8×10^{-5}	4.6×10^{-3}	4.6×10^{-3}
1999	1.1×10^{-4}	4.7×10^{-3}	4.8×10^{-3}
2000	1.6×10^{-4}	9.5×10^{-4}	1.1×10^{-3}
AVERAGE	1.8×10^{-4}	4.9×10^{-3}	5.1×10^{-3}

The collective effective dose normalised per unit of electric energy generated is presented in Table 15.12, calculated with population data up to a radius of 2000 km from the nuclear power plant.

Table 15.12 – Regional normalised collective effective dose for Embalse

YEAR	Due to gaseous discharges [man Sv $(GW_{(e)}\ y)^{-1}$]	Due to liquid discharges [man Sv $(GW_{(e)}\ y)^{-1}$]	Total Dose [man Sv $(GW_{(e)}\ y)^{-1}$]
1984	4.2×10^{-4}	1.9×10^{-2}	1.9×10^{-2}
1985	1.1×10^{-2}	3.5×10^{-2}	4.6×10^{-2}
1986	3.1×10^{-2}	2.5×10^{-1}	2.8×10^{-1}
1987	2.5×10^{-3}	2.5×10^{-1}	2.5×10^{-1}
1988	1.7×10^{-2}	2.1×10^{-1}	2.3×10^{-1}
1989	7.7×10^{-3}	2.8×10^{-1}	2.9×10^{-1}
1990	1.6×10^{-2}	2.5×10^{-1}	2.6×10^{-1}
1991	2.9×10^{-2}	5.6×10^{-1}	5.9×10^{-1}
1992	1.1×10^{-2}	1.9×10^{-1}	2.0×10^{-1}
1993	7.6×10^{-3}	2.0×10^{-1}	2.1×10^{-1}
1994	6.6×10^{-3}	1.4×10^{-1}	1.5×10^{-1}
1995	6.4×10^{-3}	2.9×10^{-1}	2.9×10^{-1}
1996	6.6×10^{-3}	3.1×10^{-1}	3.1×10^{-1}
1997	4.5×10^{-3}	1.7×10^{-1}	1.7×10^{-1}
1998	4.3×10^{-3}	2.1×10^{-1}	2.1×10^{-1}
1999	6.2×10^{-3}	1.4×10^{-1}	1.5×10^{-1}
2000	1.6×10^{-2}	3.0×10^{-2}	4.6×10^{-2}
AVERAGE	1.1×10^{-2}	2.1×10^{-1}	2.2×10^{-1}

The average collective effective dose per unit of electric energy generated, calculated with population data up to a radius of 2000 km from Embalse nuclear power plant, for the period 1984-2000, represented about 2% of the collective effective dose constraint per unit of electric energy generated.

The average collective effective dose per unit of electric energy generated due to radionuclides of global distribution, was less than 0.1 [man Sv $(GW_{(e)}\ y)^{-1}$] for tritium in the period 1984-2000 and 20 [man Sv $(GW_{(e)}\ y)^{-1}$] for C-14 in the same period. Those collective effective doses correspond to the truncate effective dose commitment integrated over the expected duration of the practice (500 years).

The average normalised effective collective dose for the period 1984-2000, 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.

15.3 Occupational Exposure

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

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

The Regulatory Body requires that whenever possible, radiological protection shall be achieved using installation systems rather than operational procedures.

Each nuclear power plant operating license sets the following conditions for workers:

1. Personnel working in a controlled area must be submitted to individual monitoring and annual medical surveillance.
2. It must be monthly recorded occupational dose due to:
   - External exposure.
   - Intake of radioactive material in this period.
3. These records must contain the following information:
   - Individual dose.
   - Collective effective dose resulting from the development of different maintenance, repairing and operation tasks.
4. The Primary Responsible must keep the mentioned records for at least thirty years after the end of service of the involved personnel.

15.3.1 Dose Limits to Workers

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

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

and

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

where:

$H_p(d)$	is the personnel equivalent dose at a depth of 0.07 mm and 3 mm (for skin and crystalline respectively), integrated in a year,
L_{DT}	is the limit of equivalent dose in skin or the lens of the eye
$H_p(10)$	is the personnel equivalent dose at a depth of 10 mm from the skin surface integrated in one year,
I_j	is the incorporation value of nuclide *j* during a year,
$I_{L,j}$	is the annual intake limit for nuclide *j*, resulting from the division of 20 mSv by the dosimetric factor of effective dose commitment for workers, per unit incorporation of the mentioned radionuclide.

15.3.1.1 Occupational dose in Atucha I nuclear power plant

In Atucha I nuclear power plant, Co60 deposits and activated corrosion products contribute with more than 60% to the occupational dose due to external exposure. For this reason, the Regulatory Body has forbidden the use of cobalt alloys in the primary circuit components. Consequently, a total replacement of fuel element channels is being carried out in Atucha I nuclear power plant (see Section 6.2.1.3) and in Atucha II nuclear power plant, now at the construction stage, there is no use of cobalt alloys in primary circuit components. This is a clear example of operational experience feedback.

The collective effective dose, the normalised collective effective dose and the average effective dose received by workers in Atucha I nuclear power plant during the period 1974-2000, are presented in Table 15.13.

TABLE 15.13 – OCCUPATIONAL DOSE IN ATUCHA I

YEAR	COLLECTIVE EFFECTIVE DOSE (MAN SV)	NORMALIZED COLLECTIVE EFFECTIVE DOSE [MAN SV (GW(E) Y)-1]	AVERAGE EFFECTIVE DOSE (MSV)
1974	1.6	15	4
1975	1.8	6	5
1976	2.3	8	8
1977	10.5	53	18
1978	5.0	15	12
1979	6.9	23	13
1980	11.5	41	16
1981	6.5	20	14
1982	12.3	41	21
1983	5.8	20	16
1984	3.2	9	10
1985	5.7	18	16
1986	8.1	25	21
1987	18.6	108	20
1988	7.9	81	14
1989	13.2	-	19
1990	10.3	48	15
1991	6.3	19	12
1992	14.9	55	14
1993	11.4	39	14
1994	8.2	27	8
1995	3.5	11	6
1996	9.7	39	10
1997	3.1	9	7
1998	6.3	22	8
1999	10.5	62	11
2000	12.0	59	16
AVERAGE	8.0	34	13

The increase of collective doses in Atucha I nuclear power plant are due to the implementation of the backfitting program (see Article 6), in particular the activities performed on the reactor internals.

15.3.1.2 Occupational dose in Embalse nuclear power plant

The collective effective dose, the normalised collective effective dose and the average effective dose received by Embalse nuclear power plant workers during the period 1984-2000 are presented in Table 15.14.

Occupational doses in Embalse nuclear power plant are lower than those recorded in Atucha I nuclear power plant due to the before mentioned contribution of Co 60, to the technological differences between both nuclear power plant as well as to the longer operation period of Atucha I nuclear power plant compared to Embalse nuclear power plant.

Table 15.14 – Occupational Dose in Embalse

YEAR	Collective Effective Dose (man Sv)	Normalized Collective Effective Dose [man Sv $(GW_{(e)}\ y)^{-1}$]	Average Effective Dose (mSv)
1984	1.0	3	1.3
1985	0.7	1	1.3
1986	2.7	7	4.4
1987	1.2	2	2.5
1988	1.9	3	5.9
1989	3.4	6	6.4
1990	1.1	2	2.2
1991	2.9	5	4.7
1992	2.4	4	3.5
1993	1.7	3	2.2
1994	0.6	1	1.1
1995	3.9	8	4.8
1996	1.2	2	2.1
1997	2.4	4	3.1
1998	2.4	4	3.2
1999	0.4	1	0.7
2000	2.9	6	3.6
AVERAGE	1.9	3.7	3.1

15.4 Regulatory Control Activities

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 personal 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 labs 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 Dosimetric 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 labs 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.

ARTICLE 16
EMERGENCY PREPAREDNESS

16.1 Introduction

The Emergency Preparedness has been explained in detail in the first National Nuclear Safety Report. On November 1998, the Executive subscribed Decree 1390 defining the scope and procedures facilitating the enforcement of the Act No 24,804, or National Law of Nuclear Activities, approved on April, 1997.

Radiological criteria regarding emergency response and preparedness remain as informed in the previous report. However, the application of Decree 1390 caused a change in the Regulatory body functions, powers and duties related to radiological off-site aspects concerning nuclear emergencies preparedness, training and response.

The main changes are:

- ✓ Regulatory Body duty of approving nuclear emergency contingency plans at municipality, state or national level.
- ✓ Regulatory Body role change from advising to leading of the off-site countermeasures implementation to mitigate radiological consequences on public from nuclear accidents.

These changes and their consequences on the organizations will be detailed below.

16.2 Regulatory Body Functions Related to Radiological Emergencies

16.2.1 Regulatory Body as a technical governmental institution

Regulatory Body role related to radiological emergencies is set by Act No 24,804 and Decree 1390.

Article 7, Decree 1390 indicates that Regulatory Body is empowered to regulate and control nuclear activity as well as to advise the Executive on issues under its purview.

Article 8, Decree 1390 defines regulation and control objectives. Item “a” defines the scope related to protection against harmful effects of ionising radiation. Furthermore the Regulatory Body shall watch over radiological and nuclear safety in the activities developed in Argentina (item “b”).

These two Articles include either normal operation or accidental situations leading to nuclear and radiological emergencies, in fixed or mobile installations, during transport or at nuclear power plants.

This inclusion is explicit in the enforcement of Article 16, item “o”, especially dedicated to nuclear accidents and emergencies.

The enforcement 1390 says as follows:

A. So as to improve the fulfilment of its functions, Regulatory Body shall approve the contingency plans for nuclear accidents, programs to face emergencies and training of nuclear plant staff and members of neighbour public. Those plans must foresee an active participation of the entire community.

B. Security Forces (Police, Gendarmerie and Naval Prefecture) and Representatives of Civil Institutions (Civil Defence, Firemen, Hospitals, etc.) within the area covered by plans shall report to the designated Regulatory Body officer…

C. The local, state and national authorities shall comply with the guidelines and criteria defined by the Regulatory Body…

Summing up:

The following conclusions result from 24,804 Act and 1390 enforcement Decree:

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

16.2.2 The Regulatory Body as nuclear power plants regulator

The Regulatory Body goes on dealing with prevention of accidents at nuclear installations from its regulatory function.

The criteria that must be adopted by nuclear power plants operators were set at the document called “Criteria for the intervention in nuclear emergencies with off-site radiological consequences”. Its compliance has been required to Atucha I and Embalse nuclear power plants. This document is being updated regarding the mentioned Act.

The nuclear power plants periodically perform drills and exercises which are evaluated by the Regulatory Body.

16.2.3 The Regulatory Body Nuclear Emergency Response System (NERS)

In order to accomplish what is set by 24,804 Act and 1390 enforcement Decree the Regulatory Body- Nuclear Emergency Response System has been created by Regulatory Body Resolution Nº 25/99 on November, 1999.

The Nuclear Emergency Response System is the organizational scheme that the Regulatory Body uses to respond in cases of nuclear emergencies and interact with the national, state and local response organizations to manage effectively nuclear emergencies at preparedness, intervention and recovery stages.

The Regulatory Body - Nuclear Emergency Response System is joined to other organizations such as: National Emergency Cabinet, States Civil Defence and Local Civil Defence of every Municipality within 10 km around each nuclear power plant.

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

A Regulatory Body Emergency Control Centre was mounted at the Headquarters in order to co-ordinate the Nuclear Emergency Response System. This centre also operates as the “Convention on Early Notification of a Nuclear Accident” and “Convention on Assistance in the case of a Nuclear Accident or Radiological Emergencies”, National Warning Point according to the International Atomic Energy Agency - Emergency Notification Assistance Technical Operations Manual (ENATOM).

16.2.4 Structure of the Emergency Plan at National Level

The Regulatory Body 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 Police, Gendarmerie, Naval Prefecture, Meteorological National 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 National Act was sanctioned.

As it was stated in the First National Nuclear Safety Report, there was not a contingency plan at national level but in 1998 there was a Project for a “National Civil Protection Program” which included nuclear emergencies. Even when this project was not concluded it stated the basis for the present Federal Emergency System which is described below.

Towards late 1998, 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. Regulatory Body and National Atomic Energy Commission are members of the Federal Emergency System.

On October 1999 the Executive sanctioned Decree 1250 establishing the Federal Emergency System.

The Federal Emergency System does not replace the function of any other response organization at different levels, ensuring that there is not overlapping responsibilities during an accident. However the Federal Emergency System is the highest co-ordinating 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. At present the National Contingency Plan for Nuclear Emergencies is being developed together by Regulatory Body and Federal Emergency System staff and is thought to be concluded by the end of 2001.

16.2.5 Nuclear Emergency Plan at Municipality Level

The emergency plans of both Atucha I and Embalse nuclear power plants are compatible with those of every Municipality within 10 km around each nuclear power plants. These plans are 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.

During last Atucha I emergency drill, the new functions, structures and organizations to respond to emergencies according to the present Act were tested. The final goal was to define the contents of the Plans according to the present legal framework, taking into account the integrated planning concepts.

16.2.6 Nuclear Emergency Plan at State Level

The Regulatory Body, the Operator and the Local Civil Defence have kept on meeting with the State Civil Defence of those States where nuclear power plants are located (Buenos Aires and Córdoba), and they participate in the exercises. The Nuclear Emergency Plans at State level will be updated according to the present Act during 2001.

ARTICLE 17
SITING

1.7 Site Re-Evaluation

As indicated in the First National Nuclear Safety Report, Article 17, the selection of both the operating (Atucha I and Embalse) and the constructing (Atucha II) nuclear power plant sites, were performed during the sixties and the early seventies and the design basic parameters used to protect such installations from external events were defined according to the criteria and the information available at that time.

Nowadays, there are more accurate tools to define these design parameters than those ones 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 events hazard, 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. The design parameters related with some of the external events included in the list were considered as follow:

- ✓ Implementation of an updated program of plant response to the earthquake occurrence at Embalse Nuclear Power Plant site was performed.
- ✓ Some studies were carried out related with both the evaluation of the tornadoes occurrence probability in Atucha I, Atucha II and Embalse Nuclear Power Plant sites; and the evaluation of tornadoes and severe storms occurrence probability in critical stretches of the external power supply to Atucha I, Atucha II and Embalse Nuclear Power Plants.

- ✓ Training and update in specific topics of man-induced events were performed. Besides, the training and final agreement with the national organization to perform this task were finished and the first activities for data collection are being scheduled. A course on man induced events due to explosions from solid substances and mitigation measures was carried out.

17.1.1. Atucha I Nuclear Power Plant:

A list of external events for re-evaluation was prepared and they were prioritised in accordance with their safety impact on both the original design and the operating experience, as follows:

- ✓ Extreme meteorological phenomena (tornadoes and severe storms)
- ✓ Man-induced events (mainly, explosions and fires, external to the plant site).

The following actions related with each external event considered 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 on December 1998. As a result of the development of the work plan, the final report draft is ready for reviewing. 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). The results obtained are presently being reviewed.
- ✓ In relation to the re-evaluation of potential hazards from *man-induced events* at the plant site, as a part of regulatory follow up actions, the training of experts from the national organization in charge of this activity was continued and the work plan was prepared.

17.1.2.Embalse Nuclear Power Plant:

A list of external events for re-evaluation was prepared and they were prioritised in accordance with their safety impact on both the original design and the operating experience, 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 related with each external event considered were carried out:

- ✓ Collection and analysis of related documents and reports.
- ✓ Regarding the re-evaluation of the operational response in case of an earthquake occurrence and the verification of the DBE excedence, two regulatory requirements were issued in 1999, regarding the implementation of an updated program of plant response to the 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 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). The equipment was installed at the control room during the 2000 planned outage which required special training for control room operators. It was agreed between the Responsible Organization and the Regulatory Body that the new USA-NRC regulatory guides would be used as a reference.

- ✓ Preparation and implementation of 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).
- ✓ Regarding the seismic risk evaluation, these requirements cover the following aspects:
  - Update and improve the program of plant response against seismic events.
  - Need to perform a seismic Probabilistic Safety Assessment.
- ✓ 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 on 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). The results obtained are presently being reviewed.
- ✓ 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. Besides, the final agreement with the national organization for performing this task was finished and the first activities for data collection are being scheduled.

ARTICLE 18
DESIGN AND CONSTRUCTION

The aspect related to design and construction of both Atucha I and Embalse nuclear power plants and the construction of Atucha II nuclear power plant were detailed in the first national nuclear safety report. From the first review meeting few activities towards the construction of Atucha II have been carried out.

18.1 Atucha II Nuclear Power Plant

The plant construction activities are still discontinued, keeping the maintenance of the components meanwhile decisions are expected to be taken so as to continue with the construction of the plant.

Nevertheless, during the last three years some activities have to be highlighted:

A. Assembling of the pressure vessel, the four moderator pumps, the moderator heat exchangers and devices to make the corresponding inspections to the pressure vessel surveillance system.

B. Commissioning of the ventilation system corresponding to both the Control Room and the rooms containing the turbine generator and electricity distribution rods switches for its own consumption.

C. Commissioning of the reactor Building and Turbine Building crane.

D. Components Storage (for example main turbine) in the rooms where they will be assembled, when they were almost conditioned to normal operation.

18.2 Regulatory Activities

Since 2001, considering that the construction might be continued, Regulatory Body increased its regulatory activity, mainly concerning the following tasks:

1. verification of the transference of the operative experience of Atucha I nuclear power plant to the design of Atucha II nuclear power plant.
2. Verification of the maintenance of the components stored within the site, the ones that are already installed or the ones that are in service.
3. Update of the applicable Regulatory Body standards, taking into account safety evolution after 1979 when Atucha II nuclear power plant construction was decided.
4. Making use of the Brazilian Regulatory Authority experience in Licensing Angra II Nuclear Power Plant, taking into account that the designer is the same as for Atucha II Nuclear Power Plant and that its construction was also stopped for a long period.

ARTICLE 19
OPERATION

The aspects related to the operation of Atucha I and Embalse nuclear power plants were described in detail in the First National Nuclear Safety Report. However, some further progress has been made in the area of operating experience feedback both from our own and from foreign plants.

19.1 Operational Experience Feedback

During the last years, the Regulatory Body examined the effectiveness of operational 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 actions follow-up.

As a consequence of a requirement issued by the Regulatory Body in 1998, the Responsible Organization started a formal and systematic process of evaluation of the operating experience in order to obtain a feedback to improve reliability and availability of the nuclear power plants.

The Responsible Organization 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 nuclear power plants. The major actions required by the Regulatory body 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 events occurrence or recurrence.
- ✓ Corrective actions follow up.
- ✓ Lessons learned from analysis.

The Responsible Organization constituted three working groups: two within each plant site and the third within the Responsible Organization headquarters, which main objectives are 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 Responsible Organization and presented to the Regulatory Body 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 nuclear power plant coming 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 Responsible Organization prepares a quarterly report including the results obtained by the application of the program. Besides, nuclear power plants expertise teams evaluate “low level events” and “near misses” obtaining their own database.

The Responsible Organization working group performs a screening analysis using international databases selecting the applicable events to domestic plants. After screening, those events are analysed in detail and they are presented to the Regulatory Body for evaluation. The evaluation could include recommendations, proposed design modifications, changes in procedures and training courses for operators if necessary.

19.1.1 Feedback of the National Operational Experience

Atucha I Nuclear Power Plant

There are many recovery actions coming from the feedback of the National Operational Experience in the case of Atucha I. Many of them are being used in the backfitting program (see Article 6).

- ✓ Loss of hydrogen in the turbine building during generator filling activities: it was decided to move the hydrogen plant outside the turbine building. At the moment this modification is still being performed. However, until modification finishes, generator fill- up operating procedure is being used.

Embalse Nuclear Power Plant

- ✓ Safety relief valves opening belonging to moisture heater / separator:
  Preventive maintenance frequency changes to avoid miscalibration.

- ✓ Load reduction due to line isolating switch:
  Review and improvement of some preventive maintenance activities.

- ✓ Calandria front fan failure:
  Recommendation was made to analyse the convenience of replacing the motor / fan support basement belonging to the entire fan.

- ✓ Loss of generation due to spurious protection activation belonging to switchgear panel:
  Preventive maintenance improvement was performed.

- ✓ Loss of power of a 220 VAC safety bus bar:
  Improvement of maintenance activities related with electronics components.

- ✓ Power reduction due to SDS #1 bar drop:
  Replacement of all the electrical resistors of each logic shutdown system #1 bar actuation module.

- ✓ Control computer malfunctioning due to analogical input card fail:
  Anomalous analogical input card replacement. As a result of the failure analysis performed it was recommended that convenience of implementing an analogical input redistribution to different analogical input chassis should be evaluated.

19.1.2 Feedback of the Operational Experience from Foreign Nuclear Power Plants

Atucha I Nuclear Power Plant

- ✓ Leakage in the Residual Heat Removal System, IRS – Civaux NPP (France): additional in-service inspection at different elbows and other accessories located near the outlet heat exchangers of the Residual Heat Removal System were decided.

- ✓ Electrician receives burns from 480 VAC arc / flash event (Callaway NPP - IMPO OE9955): applicable work procedures, including specific instructions related with personnel protection equipment use were improved.

Embalse Nuclear Power Plant

- ✓ "Level two impairment of shutdown system #2 liquid poison injection tank three on high level" (Pickering B, Unit 6 – COG):
  Modifications of the inspection program and the preventive maintenance program were implemented.

- ✓ "Tank dousing system tripped" (Gentilly-2 – COG):
  Fire protection spray valve trip affected the turbine oil tank. Actions were taken to protect different components like level sensors and engines from accidental system operation.

- ✓ Failure to test a sealed door (Gentilly-2 – COG):
  The test time used to determine penetrations and airlocks during containment leakage tests was reviewed.

- ✓ Emergency Core Cooling System fault caused by a failed mercury wetted relay and the discovery of the failure to replace three Emergency Core Cooling critical mercury wetted relay:
  Recommendations to replace all mercury-wetted relays related with the safety systems were performed.

- ✓ Degradation of pre-stressing tendon systems in pre-stressed concrete containment (NRC main generic):
  Improvements related with inspection practices. In particular, an implementation of a more detailed record of inspection findings was decided.

- ✓ "Moderator cover gas D2 excursion" (Point-Lepreau – COG):
  Heavy water conductivity limits were included to the moderator systems to be applied at the start-up after a scheduled outage where purification system isolation was performed.

- ✓ "Failure in the moderator cover gas O_2 addition line resulting in personal injury" (Darlington NGS, COG):
  An inspection and a verification procedure has been added to avoid the event occurrence.

- ✓ "Horizontal Ion Chamber Cable Connectors" (Bruce B – Candu Owner Group):
  As a part of the preventive maintenance activities related with the shutdown system # 2, connectors verification were performed.

19.2 Accident Management and Severe Accidents

Domestic nuclear power plants have a set of procedures to be applied to accident management in the two plants in operation as it was detailed in the first report. In particular, in Embalse nuclear power plant, the Responsible Organization is using the developments performed by the designer (AECL) related with generic CANDU 6 accident management through the participation in co-operation projects.

The Responsible Organization had completed a containment modelling to be applied to both nuclear power plants Atucha I and Embalse considering the hydrogen deflagration consequences.

In order to improve accident management measures taken up-to-now, in 2001 the Regulatory Body required the Responsible Organization to implement at each nuclear power an "Accident Management Program" that should aim at both prevention and mitigation by limiting damage to fission product release barriers. It should consider the different actions of damage limitation that did not succeed in preventing a rupture of the primary circuit and the containment through some pre-planned actions for recovering from the situation and limiting the release from the containment to the environment.

On the other hand, the Regulatory Body had started to analyse accident management and selected severe accidents related with both Atucha I and Embalse nuclear power plants. The activities performed are as following:

Atucha I nuclear power plant:

Hydrogen generation, transport, distribution and combustion/explosion affecting containment:

The overall stages of the Atucha I nuclear power plant containment fault as a hydrogen explosion consequence were analysed in a frame project, which is an agreement between the Nuclear Regulatory Authority and the National University of Cuyo (UNC). Besides, studies to identify the subject, to understand the associated phenomenology and to decide more detailed and specific studies were begun in Atucha I nuclear power plant. To perform the above mentioned activities some codes and models applicable to Atucha I nuclear power plant were implemented. These studies have not been completed yet and the next steps are already defined. The activities performed were as follows:

- ✓ Models using CONTAIN Code,
- ✓ Analyses of the failure modes of the Atucha I containment,
- ✓ Analyses of both pre and post accidental alternatives of inertization (as a possible risk mitigation measures),
- ✓ Severe accident modelling by using MELCOR Code,
- ✓ Dynamic loads containment's response,
- ✓ Studies related with the use of igniters and catalytic recombinators.

Embalse nuclear power plant:

Accidents beyond the design basis in CANDU 6 reactors were not included in the Final Safety Report and it implicates a core damage contribution. The particular CANDU 6 core configuration has important heat sinks available that could mitigate a severe accidental sequence when it is initiated. This heat sinks are composed by the moderator bulk inside the calandria which is surrounded by light water used in the end shield system.

At present the following works are on going :

- ✓ CANDU severe accident phenomenology assessment;
- ✓ Analysis of hydrogen explosion risk mitigation techniques using active and passive devices;
- ✓ Containment pre-stressed concrete performance studies: this work was divided into the following parts:
    - Expected loads as a hydrogen combustion and / or detonation using a CONTAIN code modelling and;
    - Containment structure response through the Sandia National Laboratory pre-stressed concrete containment vessel experiments.

As a part of the accident management program, next steps related to a detailed review of the probabilistic safety assessment (PSA) were initiated for the accident scenarios that would lead to core damage, procedures for accident prevention and mitigation will be reviewed and consequently and analysis of the necessary design changes will be performed.

In the unlikely event that a severe accident occurs it is important that the operators quickly diagnose it as such. The bulk of the severe accident research is directed towards increasing the knowledge about the severe accident phenomenology.

The hydrogen deflagration/detonation issue was identified as a major contributor to containment early failure in Atucha I, and specific studies on mitigation technologies are being analyzed; those analysis involve pre and post inertization, and use of igniters, use of recombiner and the dual method (igniters and recombiners).

As part of the implementation of Accident Management the operators´ training is being carried out covering the information obtained from the analysis related to plant specific issues which, have a significant effect on accident management decisions. Plant layout and geometry, the capacity and redundancy of emergency plant systems, as well as specific balance of plant features can determine whether a particular decision is feasible or if it makes sense under a certain accident scenario. Besides training is concentrated on plant walk-through and the plant emergency organization.

19.2.1 Performance evaluation of confinement function

Due to the fact that Atucha I is a unique plant much more effort in analyzing the safety issues has been made, in comparison to Embalse (standard CANDU 6), which has many reference plants. In relation to containment performance the studies underway have started with a qualitative analysis about the possible containment failure modes in order to improve the containment knowledge under representative severe accident condition.

Assessments of plant response to severe accidents have enabled to identify the phenomena, which may threaten the containment integrity such as overpressure due to hydrogen combustion. This analysis has allowed to know that in the case of an accident involving severe core damage by overheating, it must be assumed that large quantities of hydrogen may be produced. During the in-vessel phase about 500 kg of hydrogen provided by the oxidation reaction between Zircalloy and steam would occur. Besides, in the ex-vessel phase a hydrogen additional generation is expected by corium-concrete interaction.

Structural containment capacity under both static and dynamic loads due to severe accident condition were reviewed about the strategy and general methodology that can be used to quantify the ultimate performance of the containment. The main contribution of these analysis is the better knowledge about the specific features of the containment performance. As an example the containment weaknesses of Atucha-I zones of spherical shell in the neighborhood of reinforced sections have been identified. Additionally, to support severe accident research the participation in international R&D programs such as CSARP (Cooperative Severe Accident Research Program) is being carried out.

19.3 Peer Reviews and Activities between the Responsible Organization and WANO from 1998 to 2001

The Responsible Organization (Nucleoeléctrica Argentina S.A. -NASA) is a member of the World Association of Nuclear Operators (WANO) created in 1988. First through the Atomic Energy National Commission and then ratified, in Paris 1995, its condition of associated as NASA.

Both nuclear power plants under operation received WANO Peer Review Missions. NASA participated in all the WANO programs through the WANO – PC (Paris Centre), the following being relevant interactions:

A. Technical Exchange Visits:

- ✓ Technical visit Embalse nuclear power plant experts to Point Lepreau nuclear power plant – (Canada).
- ✓ Technical visit of Atucha I nuclear power plant experts to Borsele nuclear power plant – (the Netherlands).
- ✓ Technical visit of Wolsong I nuclear power plant (Republic of Korea) experts to Embalse nuclear power plant – (Argentina).

B. b) Peer Review Missions

Argentina has provided different specialists from both nuclear power plants to participate in Peer Review Missions.

- ✓ Contribution of specialists from Embalse nuclear power plant:
  - Peer Review training at Hunterstone nuclear power plant, United Kingdom, May 1999;
  - Peer Review at Chapelcross nuclear power plant, United Kingdom, September 1999;
  - Peer Review at Saint Alban nuclear power plant, France, November / December 1999;
  - Training for Exit Representative at South Ukraine nuclear power plant, Ukraine, June 1999;
  - Exit Meeting at Chapelcross nuclear power plant, United Kingdom, October 1999;
  - Peer Review at Kanupp nuclear power plant, Pakistan, February / March 2000;
  - Peer Review at Chapelcross nuclear power plant, United Kingdom, July 2000.

- ✓ Contribution of specialists from Atucha I nuclear power plant:
  - Peer Review training at Philipsburg nuclear power plant, Germany, April 1999;
  - Peer Review at Grafenrheinfeld nuclear power plant, Germany, January / February 1999;
  - Peer Review at Duane Arnold nuclear power plant, United States, February / March 1999;
  - Peer Review at Braidwood nuclear power plant, United States, May / June 1999;
  - Osart Follow-up Mission at Hartlepool nuclear power plant, United Kingdom, June 1999;
  - Peer Review at Hinkley Point B nuclear power plant, United States, United Kingdom, January 2000;
  - Peer Review at Kiwaunee nuclear power plant, United States, March 2000;
  - Osart Follow-up Mission at Sizewell nuclear power plant, United Kingdom, June 2000;

C. Operative Experience Exchange Program:

- ✓ - Issue of both Atucha I and Embalse Relevant Event Reports.

- ✓ - Use of the operative experience WANO database.
- ✓ - Issue of both Atucha I and Embalse Performance Indicators.

D. Participation in Peer Review Missions

Lectures and Participation in Courses and Symposiums: The specialists of NASA from Atucha I, Embalse and the headquarters have participated in the following courses and symposiums:

- ✓ “Industrial Safety”, Hinkley Point nuclear power plant, United Kingdom, May 25/26, 1999.
- ✓ “Application of PSA”. Dukovany nuclear power plant, Check Republic, March 17/19, 1999.
- ✓ “Public Relations in the local community”, Embalse nuclear power plant, Argentina, August1999.
- ✓ “Outage Review-PO and Criteria”, Paris, France, October 1999.
- ✓ “Chemistry Control”, Lyon, France, October 20/26, 1999.
- ✓ “Containment Integrity”, Osthamar, Sweden, October 25/28,1999.
- ✓ “Human Performance Enhancement”, WANO PC, Paris, France, November 8/10, 1999.
- ✓ Workshop on “Training Qualification”, Daya Bay, Guangdong nuclear power plant, China, May 26 to June 1, 2000.
- ✓ “Manager Executive Course”, WANO PC, Paris, France, September 2000.

ANNEXES

ANNEX I
CONCLUSIONS ABOUT ARGENTINA DURING THE FIRST REVIEW MEETING ON THE CONVENTION ON NUCLEAR SAFETY

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.

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.

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.

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.

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
ANSWERS TO QUESTIONS OR COMMENTS - NATIONAL NUCLEAR SAFETY REPORT - 1998

N°: 1
CNS-Ref.-Art.: 6
Page of Report: 6.2
Chapter of Nat. Report: 6.1
Country: Canada

The report states that, since the beginning of nuclear activities in Argentina, many safety-related improvements have been implemented at nuclear installations as a result of safety analyses, regulatory inspections and operational experience. Are there any major improvements to safety that still have to be implemented, and what is the schedule?

The safety improvements implemented or in progress included in the Report, as well as those occurred after its issue -explained at the end of the presentation- are all the improvements considered by the Regulatory Authority up to date.

At present, no additional improvements are foreseen. Nevertheless, due to the periodic safety improvement program mainly based on risk management using a probabilistic safety analysis, safety assessments and the operational experience feedback, some aspects might appear the importance of which is not possible to evaluate before considering the results of the corresponding studies.

N°: 2
CNS-Ref.-Art.:6
Page of Report: 2-30
Chapter of Nat. Report:6.2, 6.2.1.3, 6.2.1.4, 6.2.2.2, 6.2.2.6,
Country: Canada

The report described a number of improvement activities that are scheduled to be completed in the next few years, such as:

- ✓ **Replacement of all coolant channels for ATUCHA I by end of 1999;**
- ✓ **Installation of an additional core residual heat removal system for ATUCHA I NPP by the year 2000 (section);**
- ✓ **Inspection of all pressure tube channels for "spacer location and repositioning" for EMBALSE NPP by 2003;**

- ✓ **Plant specific probabilistic safety assessment, phase 1, for EMBALSE NPP completed by middle 1999.**

Has sufficient progress been made on these activities to meet the scheduled completion dates?

Both the National Report and this presentation in its last part have answered these questions, except for the case of Embalse's PSA.

For this last case, the corresponding schedule have been postponed in six months due to delays in the supply of thermal hydraulics codes applicable for CANDU6.

N°: 3
CNS-REF.-ART.: 6
PAGE OF REPORT: 5/6
CHAPTER OF NAT. REPORT: 6.2.1.4.
COUNTRY: GERMANY

As the plant is not designed to control small LOCAs, are there accident management measures provided to mitigate possible consequences of a small LOCA until the planned second heat sink is available?

In fact, the original design for Atucha I does not foresee the occurrence of LOCA in which there is a small breakage of the cold leg, producing a slow depressurisation.

Nevertheless, safety assessments were carried out in order to analyse the possibility of using the volume regulation system to prevent overheating of fuel elements during the period between the breakage occurrence and the moment when the low pressure emergency injection system is activated.

As a result of the PSA an operation procedure was elaborated enabling the operator to modify manually the route of injection of the volume regulation system.

The operators are specially trained to carry out such task and have enough time to do so.

N°: 4
CNS-REF.-ART.: 6
PAGE OF REPORT: 5/6
CHAPTER OF NAT. REPORT: 6.2.1.4.
COUNTRY: GERMANY

Is a modification of the reactor protection system planned in connection with the implementation of the second heat sink?

The design of the second heat sink consider its complete independence from the reactor protection system.

It consists of two identical and redundant, that include instrumentation, control and power supply trains.

Nevertheless, in connection with the implementation of the second heat sink, the originally existing reactor protection system will be modified to consider the operation of the emergency core cooling system in the range of small LOCAs.

N°: 5
CNS-Ref.-Art.: 6
Page of Report:
Chapter of Nat. Report: 6.2.1.1
Country: Germany

The Regulatory Body required...to improve the emergency power supply system..."

Does this include the improvement of the physical separation (protection against fire and internal flooding) of the emergency diesels?

As it was already explained in the Report, barriers against fire were installed between diesel generators and flammable fluids sumps, and additional improvements were carried out in the diesel generators and other components of the emergency energy supply system, with the purpose of reducing fire risk.

Concerning internal flooding, the probability of occurrence of such event in the room of the emergency diesel generators, together with a total loss of off site power, was evaluated.

Although the value of such probability is negligible, if such event occurs, the automatic actuation of redundant and independent diesel generators, provided as part of the backfitting program, is foreseen.

N°: 6
CNS-Ref.-Art.: 6
Page of Report: 6
Chapter of Nat. Report: 6.2.1.5.
Country: Germany

"During this period the electrical power needed for core heat removal ...is supplied by a hydraulic turbine...."

What are the consequences if the synchronization of the emergency generators with the hydro-generator fails?

Is a secured emergency power supply planned?

After the occurrence of a loss of off-site power, the hydraulic turbine will operate keeping the voltage in the emergency bus bars during some 40 seconds.

During such period, diesel groups will enter in service and, if they synchronise satisfactorily, they will couple to rods under voltage.

Under such low probable conditions, the emergency power supply system is effectively non-interruptible.

In case of occurrence of any perturbation or synchronization failure of diesel generators with the hydraulic-turbine a loss of power supply to emergency for some seconds will occur.

In this situation the diesel generators will be available to couple emergency buses that remain without voltage.

The safety impact to this short interruption of the emergency power supply have been taken into account in the PSA, for all the scenarios considered, concluding that the outage of moderator system pumps during the interruption is not significant from the safety point of view, due to the great inertia of the heavy water mass flowing along the primary/moderator circuit.

Besides, CNA I has the emergency power diesel generator system additionally installed as part of its back-fitting.

This system begins operating after a time period of 60 seconds without voltage in the emergency rods.

N°: 7
CNS-Ref.-Art.: 6
Page of Report: 7
Chapter of Nat. Report: 6.2.1.6.
Country: Canada

The report states that the probabilistic safety assessment of the nuclear power plant CNA I has demonstrated some weaknesses in the plant design and operation.

How are such weaknesses affecting the CNA I safety level?

The PSA enabled the detection of weaknesses in design and operation procedures, as well as the identification of different alternatives for the improvements to be carried out. To this respect, the Regulatory Body submitted a set of requirements to be immediately fulfilled.

For instance, a weakness was found concerning small LOCAs due to an eventual open failure of the pressuriser's safety valve, which was improved by providing automatic pressure control of the pressuriser.

Moreover, unforeseen situations in the original design, producing depressurisation of the primary circuit, were solved by improving the pressuriser's spray system and in the operational procedures for this kind of transients.

The original design had foreseen a unique relief route for the secondary system pressure with the reactor on power, through the safety valves of steam generators.

In order to improve the reliability in such events it has been implemented reactor trip due to high pressure in steam generators and redundancy in vapour release.

Thus, this contributed to decrease requirements of opening and possible closing failure of steam generators safety valves.

The significant contribution to risk of core damage due to transients “Loss of feedwater system due to pump and valve failures”, “Loss off site power“, “Loss of main condenser”, will be reduced by the inclusion of the second heat sink.

The contribution to risk due to a failure in the emergency power supply system was decreased installing new diesel generators for emergency power supply, which significantly improve the reliability of such system.

The most important improvements carried out in operational procedures are related to events of high and low pressure transients of the primary system, loss of feedwater and loss of cooling water from the river.

N°: 8
CNS-Ref.-Art.: 6
Page of Report: 7
Chapter of Nat. Report: 6.2.1.6.
Country: Germany

What is the frequency of core damage derived from the PSA, and what are the main contributions?

The value of the core damage probability obtained from the PSA, taking into account the improvements already carried out and in progress, is less than 10^{-04} per year. A risk management program is implemented with the objective of reducing the core damage frequency even more than the above mentioned value.

It should be mentioned that the concept "core damage" used in the 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.

Transients contribute up to 80% to core damage probability, while events of the LOCA type contribute in 20%. As concerns transients, the main contributor is the loss of main feedwater to steam generators. In the case of LOCA, the main contributor is the loss of primary coolant through the pressuriser safety valve.

N°: 9
CNS-Ref.-Art.: 6
Page of Report: 8
Chapter of Nat. Report: 6.2.1.6.
Country: Canada

The report states that, since the nuclear power plant CNA I has been operating for more than twenty years, the Regulatory Body has issued a number of requirements with the objective to increase the safety level.

What are the major requirements related to the CNA I backfitting plan, and are they related to aging?

As mentioned in the Report, the major requirements related to the CNA I backfitting plan are:

- ✓ Installation of a second heat sink.
- ✓ Installation of additional valves in the pressuriser of the primary heat transport system.

- ✓ Improvements in the shutdown core cooling system (residual core heat removal).
- ✓ Improvements in the emergency electric supply system.
- ✓ Updating of emergency operation procedures and the corresponding training of operators.
- ✓ Improvements in thermal isolation materials of fuel cooling channels.
- ✓ Improvements to the reactor pressure vessel surveillance program.
- ✓ Better estimation of safety margin for the reactor pressure vessel.

The last three items are directly related to ageing.

N°: 10
CNS-Ref.-Art.: 6
Page of Report: 8/9
Chapter of Nat. Report: 6.2.1.7.
Country: Germany

Which analyses or measures are intended to determine the safety margin of the CNA-I reactor pressure vessel?

The pressure vessel surveillance program for Atucha I has several uncertainties concerning the specimens irradiation. There are some doubts about the ductile-to-brittle transition temperature shift calculated and extrapolated for 32 full power years.

The most important sources of uncertainties come from: the significant differences in neutron spectra for the irradiated specimens, the high acceleration factor of the specimens irradiated in the VAK (German) reactor and to the different probe orientations in the diverse experiments performed.

Concerning the minimum admissible temperature for the pressure vessel material, the value obtained seems to be the most conservative one taking into account the scenarios considered for LOCA, although there are also uncertainties for this kind of studies.

The Responsible Organization is evaluating, upon request of the Regulatory Body, the set of tests and studies to carry out in order to reduce the above mentioned uncertainties.

Such studies must be based on scenarios that consider more realistic hypothesis for transients and other accidents, and must unable a complete thermal-hydraulic analysis as well as a better evaluation of fracture-mechanical parameters associated to thermal shock.

Additionally, the Regulatory Body required the implementation of necessary design changes to heat up the accumulator water of high pressure emergency injection system with the objective of improving its safety margin at the end of the plant life time.

N°: 11
CNS-Ref.-Art.: 6
Page of Report: 9/10
Chapter of Nat. Report: 6.2.1.8.
Country: Germany

Which consequences has the use of slightly enriched fuel elements for the reactor pressure vessel with respect to embrittlement?

Is there information available about changes in other essential plant parameters due to the use of slightly enriched fuel?

Concerning the first question, studies carried out show that:

- ✓ When a 100% slightly enriched uranium core is compared to a 100% natural uranium fuelled core, it is observed that the neutron flux at the inner surface of the pressure vessel wall (averaged axially and in angle) is approximately 4% greater in the first case.
- ✓ For the same two different cores, the maximum value for the neutron flux, both in axial and angular directions, in front of those channels which are nearer to the vessel inner surface is some 6% smaller in the first case.

These differences, mainly due to a different fuel management strategy, indicate that the fluence reached at the end of the reactor lifetime will not be significantly different from the one it would be achievable with a 100% natural uranium fuelled core.

As regards the second question, the safety assessments performed include :

A. The verification of safety margins for the reactor shutdown systems, and

B. The transient and accident analysis.

The results of studies indicated in a) show that slightly enriched uranium does not endanger the effectiveness of the reactor shutdown systems. In case b), for those events with positive reactivity insertion and fast power increase, with characteristics that cannot be controlled by the regulation system, such as large LOCA, the study shows that the plant transients are slightly more risky for slightly enriched uranium fuel, but even for such case the applicable acceptability limits are not exceeded.

For other events affecting process systems (such as decay heat removal) it is shown that the plant’s response is not affected by the presence of slightly enriched fuel.

Among the accidents analysed, the mistaken insertion of a slightly enriched fuel in the central position, and neutron flux oscillations due to the spatial variation of xenon concentration should be mentioned. In both cases the results obtained show that the situations are well managed by the reactor control system.

N°: 12
CNS-Ref.-Art.: 6
Page of Report: 6.2
Chapter of Nat. Report: 6.2.1.1.
Country: China

Please give details, and solution, of fuel channel event occurred in 1988.

During full power operation in August 1988 a sudden reactivity loss was detected. Power control was disturbed and the operator actuated SCRAM manually, because he interpreted the event as a control rod spurious drop.

During xenon cycle primary system remained in hot shut down. Since no damage was detected and safety systems were unaffected, the Safety Advisory Internal Committee suggested the plant manager to start up the plant again, and perform some specifics tests.

Criticality was normally obtained and the plant was again connected to the grid two days later. Since heavy water flow in moderator loop number one was reduced 15% approximately, power through moderator cooler was higher than normal, a vibration was observed in an ex-core neutron flux detector and a slight rising in gamma activity of moderator system was detected, so power was limited at 70%.

Due to the fact that gamma activity rose, the procedure followed was to look for fuel element failure during recent refuelling operations. Three days before the incident, higher sipping values at R06 channel were detected. Also, a neutron flux detector, which was under vibrations, was placed near R06 channel.

For this reason, an inspection of the last two defuelled fuel elements corresponding to the cooling channel R06 was determined. Serious freeting damages were observed.

The fuel element located in R06 position was withdrawn and placed in a decay pool. Underwater visual inspection showed that the lowest 80 cm. of the fuel element had been lost.

As a result of this evidence and the certainty of coolant channel rupture, both the plant shutdown and an inspection of its internals was decided.

The inspections using TV cameras for underwater and high radiation fields, were done passing them through the top of the cooling channels 12 cm. diameter, and immersing them in water up to about 12 meters deep. The mentioned inspection showed the following failures

- ✓ Breakage of cooling channel R06 with loss of the active lower two meters;
- ✓ Damage on the R06 neighbour cooling channel Q05;
- ✓ Breakage of the level sensor guide tube W03;
- ✓ Damage on the R06 neighbour cooling channels thermal isolating foils;
- ✓ Falling down of some moderator tank thermal isolating plates and damages in the moderator distribution lines.

Failure analysis suggests that it began around 1982/1983, starting with the W03 level measuring guide tube rupture. This guide tube once detached caused the cooling channel rupture by a slow freeting action.

Channel debris struck against moderator tank thermal isolation and several surrounding channel thermal isolation foils. There were several previous indicators that were not correctly understood summarised as follows:

- ✓ Measuring level probe failure since 1982/3.
- ✓ This level probe was withdrawn in 1987 and deposited in decay pools without realising that it was damaged.
- ✓ Moderator circuit average temperature higher than normal.
- ✓ Slight increase in fuel consumption.
- ✓ Spent fuel element damages, coming from R06, were not observed at a proper time.
- ✓ Sipping indications in refuelling machine.
- ✓ Abnormal indications in corrected neutron flux detector close to R06.

The repair work program included the following actions:

- ✓ Removal of fuel channel R06;
- ✓ Removal of the level probe guiding tube W03;
- ✓ Removal of loose parts, mainly insulating foils from damaged fuel channels and pieces of insulation plates from the moderator tank;
- ✓ Removal of fuel pellets and pieces of fuel cladding from the damaged fuel element R06;
- ✓ Repair of damaged lower moderator distribution toroid;
- ✓ Cutting and removal of level probe guiding tubes W01, W02 and W05, in which significant bending was observed;
- ✓ Removal of damaged pieces of the insulation plate, which could become loose from the moderator tank wall;
- ✓ Replacement of all the channels affected after inspection in the fuel storage pool.

The total repair operation lasted for about 16 months, and the total exposure dose to operators during this time was 8 man-Sv. Dose rate to workers above the reactor vessel was 1 mSv/h.

About 150 people were involved in the reactor internal repair. The highest doses were received by supervisors and people working on the removal and reinstallation of fuel channels.

When the repairing program was fulfilled, a safety reassessment was done taking into account the accomplishment of the follows safety functions:

- ✓ Stop of power generation (shut-off) in any condition;
- ✓ Residual heat removal and
- ✓ Monitoring of the main plant parameters.

Besides, in view of the fact that an adequate interpretation of what was happening would have limited the actions only to level probe changing and, eventually a cooling channel replacement, it was decided to emphasise the implementation of several early warning techniques through a program including the following activities:

- ✓ Periodic inspections of moderator tank internals with TV devices;
- ✓ Improvement of failure fuel element detection system;
- ✓ Nuclear instrumentation surveillance;
- ✓ Permanent power measurements;
- ✓ Visual inspections of each fuel when discharged in both cases by sipping in the refuelling machine and by means of optical devices in decay pools (on-line inspection of all fuel elements);
- ✓ Differential pressure measurement taken by the reactor refuelling machine on each cooling channel when operating in refuelling or exchanging fuels in the reactor (on-line inspection of all fuel elements);
- ✓ Neutronic noise for detecting abnormal movements of reactor internals and
- ✓ Systematic assessment of the routine test results of shut down systems and the emergency core cooling system (with special attention to the emergency cooling circulation system (shut down cooling mode).

More detailed information about the fuel channel event occurred in 1988, see “Annex 6 – IAEA Safety Review Mission at the Atucha I NPP” in the document Annexes of the National Nuclear Safety Report.

N°: 13
CNS-Ref.-Art.: 7/17
Page of Report: 7/13
Chapter of Nat. Report: 17.3
Country: Brazil

Is there any other regulatory organization involved in the licensing of a nuclear power plant in Argentina? If yes, what are the scope of the reviews?

According to the “National Law of Nuclear Activity” (Act No 24,804) the Nuclear Regulatory Authority is in charge of regulation and control of nuclear activity concerning radiological and nuclear safety, physical protection and control of the use of nuclear materials, licensing and control of nuclear installations and international safeguards; it is also in charge of advising the National Executive Power about subjects of its competence.

Besides, the above mentioned law declares that regulation and control of nuclear activities are subject to domestic jurisdiction in the already mentioned aspects, and establishes that any new significant nuclear installation must have a construction licence authorising its sitting, issued by the Nuclear Regulatory Authority with the approval of the provincial Estate in which the considered site is located.

From the above considerations it may be concluded that the Nuclear Regulatory Authority is the only domestic authority involved in the licensing process of nuclear installations; the before mentioned approval of a provincial Estate is not of technical nature.

N° 14
CNS.Ref.-Art. 7
Page of Report: 11
Chapter of Nat.Report: 7.4
Country: Brazil

What has been the experience in using sanction? How many penalities have been imposed to the Responsible Organization?

As already mentioned in the National Safety Report, the Regulatory Body considers that consensus and conviction must prevail for the better fulfilment of the requirements submitted to the licence holder of nuclear installations and their staff.

Therefore, sanction application is not a routine regulatory action but the last measure to be taken in case of conflictive situations.

Since the beginning of nuclear power plant operation in the country, it once happened that the validity of the operation licence of Atucha I was suspended, when a cooling channel broke, event already commented when answering Question No 12 as well as in the National Report.

It also happened once that a sanction was applied of non-renewal of specific authorization due to regulatory reasons.

On the other hand, during Embalse commissioning (June 1983), the Regulatory Body suspended the authorisation to proceed with normal tasks until the problems detected were solved (Stop Work during three months), due to evidences of management and quality assurance failures.

N°: 15
CNS-Ref.-Art.: 7
Page of Report:
Chapter of Nat. Report: 7.2.2.1.
Country: Australia

Section 7.2.2.1 refers to deterministic and probabilistic approaches being complementary and to be used in a balanced manner.

We would appreciate more detailed information about the practical experience of the Responsible Organization and the Regulatory Body in the application of this balanced approach and clarification of the extent to which probabilistic methods are a regulatory requirement or just an assessment tool supplementing deterministic regulatory objectives.

The practical experience in the application of deterministic and probabilistic methods in nuclear safety in Argentina may be evidenced both in standards and requirements issued by the Regulatory Body, and in safety related actions carried out by the Responsible Organization.

Concerning regulatory standards, their probabilistic contents is evidenced through reliability objectives, probabilistic criteria regarding accidental sequences and probabilistic radiological criteria for accidents.

Deterministic contents of standards are evidenced in subjects such as safety margins, operation limits and conditions and defence in depth criteria.

As an example of both probabilistic and deterministic criteria used together, a regulatory standard should be mentioned related to Instrumentation and Protection Systems.

Such standard establishes that design shall restrict the probability of undue actions of the protection system without affecting its corresponding compatibility.

As an example of deterministic criteria, Atucha I operation licence establishes that the installation shall be operated within limits and conditions established in the mandatory documentation.

As an example of probabilistic nature, the above mentioned licence sets that periodic tests of components, equipment and systems belonging to the installation shall be carried out as frequently as necessary in order to ensure that their reliability lies within design values.

Besides, the Regulatory Body requires a safety assessment of the plant by means of a probabilistic safety analysis, both in operation as during shutdown states.

Safety analyses are also required to assess reliability of components, equipment and systems. Moreover, requirements have been submitted for the improvement of safety on the basis of PSA results.

As concerns deterministic items, the Regulatory Body requires the elaboration of preliminary and final safety reports, the performance evaluation of components, equipment and systems and any other aspect related to barriers to prevent radioactive material release to the environment.

Concerning practices of the Responsible Organization, an example of them is the elaboration of the PSA to take decisions about design and operation, and the improvement of maintenance and surveillance programs (periodic tests and in-service inspections).

N°: 16
CNS-Ref.-Art.: 7
Page of Report: 3-4
Chapter of Nat. Report: 7.2.2.2.
Country: Canada

The report indicates that the regulatory system in Argentina is performance-based, and uses both the deterministic and probabilistic approaches to regulation.

How effective is the performance-based regulatory system adopted in Argentina, and what are its main strengths and weaknesses?

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 the good engineering judgement, in the operators qualification and in the Responsible Organization's appropriate way of taking decisions.

It is in such context that the Responsible Organization must convince the Regulatory Body that the installation is safe. The role of this last one is to be sceptic and critical, without proposing "how" and, at most, without finding objections to the solutions found by the first one.

The before mentioned conviction implies an interaction of intense, continuous and personal nature among professionals of the Regulatory Body and the Responsible Organization, along the installation lifetime, without affecting the independence of both institutions.

It may be thus asserted that the effectiveness of the regulatory system adopted in the country is reasonable according to the results it produced as time went by.

The probabilistic safety analyses of both operating nuclear power plants constitute a couple of recent examples regarding such effectiveness.

As concerns the adoption of a performance conception, some of the most important advantages, verified through experience in their application, are:

- ✓ The nature of the interaction between the Regulatory Body and the Responsible Organization contributes to the minimization of the possibilities of non-compliance or deficient compliance with regulatory requirements and enables time and effort saving in fulfilling such requirements.

  - The solutions to regulatory requirements come, in general, from the Responsible Organization, that is to say, from the organization that knows in detail everything concerning structures, components, equipment and systems constituting the installation involved.

  - The establishment of safety objectives helps in facing safety questions with a high degree of flexibility as compared to 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 thus be satisfactorily solved in time and form.

  - Each non-objected solution indicates a high degree of consensus between the Regulatory Body and the Responsible Organization, so that its implementation is made extremely easier.

Concerning disadvantages of the Argentinian regulatory system, up to date no significant one has been detected. It may be mentioned, however, that in some opportunities it is difficult to verify the performance fulfilment in the solutions proposed by the Responsible Organization.

N°: 17
CNS-Ref.-Art.: 7
Page of Report: -
Chapter of Nat. Report: -
Country: Spain

Regarding NPP personnel, activities related to "licensable functions" are described.

Are these functions limited to control room operators or do they include other responsibilities, i.e. radiation protection?

The licensable functions are not limited to control room operators, but they include some other responsibilities.

Annex to "Resolution No. 1054/75 – Rules for Licences and Specific Authorizations of Personnel of Nuclear Power Plants and Nuclear Reactors" establishes criteria and procedures for issuing licences and specific authorizations, as well as the terms and conditions under which the Regulatory Authority grants such licences and authorizations.

Besides, the Annex establishes that the type and number of functions shall depend on the installation's magnitude and on the operational organization chart. For the case of nuclear power plants, it defines the following functions:

1.- Director,
2.- Production Assistant,
3.- Technical Engineering Head,
4.- Operation Head,
5.- Shift Head,
6.- Primary Circuit Head,
7.- Secondary Circuit Head,
8.- Loading System Operator,
9.- Primary Circuit Assistant,
10.- Pool Assistant,
11.- Engineering Head,
12.- Mechanics Section Head,
13.- Electric Section Head,
14.- Regulation and Control Head,
15.- General Supervisor,
16.- Mechanics Supervisor,
17.- Electric Supervisor,
18.- Regulation and Control Supervisor,
19.- Preventive Maintenance Supervisor,
20.- Radiation Protection and Safety Head,
21.- Radiation Protection Head,
22.- Radiation Protection Supervisor,
23.- Radiation Protection Officer.

It should be added that the Regulatory Body carries out a control of evaluations to non licensed personnel at he installation.

N°: 18
CNS-Ref.-Art.: 7
Page of Report: 7-5
Chapter of Nat. Report: 7-3-2
Country: Canada

The report describes the licensing process for nuclear installations and their personnel.

Is there a practice of public information and input into the regulatory licensing process? If yes, please describe how useful it is, and its possible impact on the licensing process.

No legal disposition exists demanding the Regulatory Body to carry out routine practices of public information during the licensing process of a nuclear installation.

Nevertheless, if any public consultation or request of additional information to the Regulatory Body occurs, be it personal, legal or through the People's Defender, the Regulatory Body answers any of them related to subjects of its competence.

On the other hand, the National Law of Nuclear Activity demands that any new siting for a nuclear installation must be approved by the province involved.

It is a decision of the provincial authorities to carry out a public consultation.

This process is independent of installation licensing carried out by the Regulatory Body.

N°: 19
CNS-Ref.-Art.: 7
Page of Report: 9-10
Chapter of Nat. Report: 7.3.3.
Country: Canada

The report states that both routine and special inspections are performed by the regulator. They are prioritised according to plant specific probabilistic safety assessment results.

Please provide indications of the frequency of regulatory inspections and the type of deficiencies they uncover.

Routine inspections are daily carried out by resident inspectors of nuclear power plants, and they correspond to the kind of planned inspection.

They are based on pre-established procedures having specified purposes, defined functions, competence and activities of resident inspectors.

Different areas and subjects to be inspected are also specified as well as their importance concerning safety.

On the other hand, special inspections are specific evaluations carried out by resident inspectors, together with specialists in different fields of radiological and nuclear safety.

These inspections are carried out in response to unexpected situations or events, abnormal or unusual ones, be them announced or not. Their frequency is variable because they depend on the installation performance.

The purpose of regulatory inspections is to verify compliance with what is established in the mandatory documentation, and particularly in the operation licence.

They include verifications related to normal operation limits and conditions, surveillance program, periodic tests, preventive and predictive maintenance program and occupational radiological protection.

Usually, the inspection effort is increased during programmed outages in order to cover special tasks carried out during such outages, as the in-service inspection program and preventive and corrective maintenance.

The use of PSA during regulatory inspections represents an additional tool for inspectors that helps in a better assignment of resources and priorities.

The aspects taken into account for such assignment are: importance of components, equipment and systems related to their impact on safety, priority given to equipment and systems related to the occurrence of initiating events, paying particular attention to those components, equipment and systems that participate in the most significant sequences related to damage to the core frequency.

Major deficiencies/anomalies detected by regulatory inspections:

- ✓ Execution frequencies of safety related systems preventive maintenance.
- ✓ Execution frequencies of safety related systems surveillance tests.
- ✓ Renewal process of specific authorisations related to operation personnel.
- ✓ Pressure tubes in service inspection practices during scheduled outages.
- ✓ Reduction of established training period.
- ✓ Spare parts procurement related to scheduled outage activities.
- ✓ Execution of Call-ups maintenance activities.
- ✓ Surveillance tests execution without using forms.
- ✓ Maintenance reworking due to human errors.
- ✓ Planning and execution of scheduled outage activities.

N°: 20
CNS-Ref.-Art.: 8
Page of Report: 4
Chapter of Nat. Report: -
Country: Brazil

What is the policy related to resident inspectors? Do they rotate from time to time between site and headquarters?

The policy of the Nuclear Regulatory Authority concerning resident inspectors work is such that they stay at the installation during an approximate period of six to eight years, being then transferred to the headquarters to proceed with technical tasks related to the inspection function itself.

N° 21
CNS.Ref.-Art. 8
Page of Report: 9
Chapter of Nat.Report: 8.4
Country: Canada

The report indicates the independence of the regulatory body with respect to its relationships with other organizations.

Please describe the independence of the regulatory body in relation to its position in the government structure.

The Regulatory Body, officially named Nuclear Regulatory Authority, is an administrative and financially independent organization directly reporting to the Presidency of the Republic, that is to say, to the maximum government stage of the country.

On the other hand, the company named Nucleoléctrica Argentina S.A., the operation licence holder, now being under the process of privatisation, depends, for the time being, on the Ministry of Economy and Public Works and Services of the Nation.

The National Atomic Energy Commission, in charge of establishing policies, and of research and development in the field of atomic and nuclear applications, is also under the jurisdiction of the Presidency of the Republic.

N°: 22
CNS-Ref.-Art.: 8
Page of Report: -
Chapter of Nat. Report: -
Country: Spain

It is stated that ARN is funded by the Government and 20% of its total expenditure is collected through fees.

Are there plans to accommodate the fees to real expenditures?

There are no plans to modify the relationship between funds from the government and incomes from regulatory resources.

The reason for this attitude is the ARN must be completely independent from the Responsible Organization, as well as from users of nuclear energy in any possible form, including from the financial point of view.

N°: 23
CNS-Ref.-Art.: 9
Page of Report: -
Chapter of Nat. Report: 9
Country: Spain

Mention is made of the financial responsibilities of the operator for potencial damages to the public or the environment.

Has the ARN any review responsibilities before granting the licence?

The company in charge of the operation of nuclear installations in the country (NUCLEOELÉCTRICA ARGENTINA S.A.) is presently under the process of privatisation.

Therefore, and while nuclear power plants are operated by a company owned by the National State, no insurance or warranty are needed to cover the damages associated with the eventual occurrence of a nuclear accident, in the terms expressed in the VIENNA CONVENTION ON CIVIL RESPONSIBILITY FOR NUCLEAR DAMAGES, because the State will always be in charge of them.

From the moment of effectiveness of the privatisation, and according to what is established in Act No 24,804 "National Law of Nuclear Activity" in its Article 9°, the operator of a nuclear power plant will be compelled to assume the civil responsibility up to an amount of 80.000.000 dollars for each nuclear installation, as the already mentioned Vienna Convention establishes.

Such amount of money shall be covered by an insurance or a financial warranty to the National Executive Power's satisfaction, or the organization it appoints, assuming the National State the remaining responsibility.

From the above mentioned arguments it may be concluded that it is not the ARN responsibility to take any action prior to granting a licence.

N°: 24
CNS-Ref.-Art.: 9
Page of Report: 2-4
Chapter of Nat. Report: 9.2
Country: Canada

The report describes the Responsible Organization for an NPP and its main responsibilities.

What are the responsibilities and related activities of the Responsible Organization in the following areas: safety analysis reviews, reliability studies, peer evaluations, and the arrangement of external evaluations.

As it is mentioned in the National Safety Report, the Regulatory Body establishes in the operation licence that a Technical Review Committee must exist independent of the installation, giving advice to the highest positions of the Responsible Organization.

Its functions are:

- ✓ Review the development of plant operation.
- ✓ Analyse the importance of foreseen operational incidents and initiating events occurred.
- ✓ Evaluate proposals of design modification that could affect safety of the installations.
- ✓ Analyse the minutes of the Safety Advisory Internal Committee for each plant.
- ✓ Review modifications in the mandatory documentation.

On the other hand, as part of NASA's organizational chart, the Nuclear Regulation, Safety and Safeguards Management Section (located at its headquarters) carries out safety assessments, reliability studies, risk analyses and other tasks such as updating of mandatory documentation. Besides, this section centralises the relationship between the Responsible Organization and the Nuclear Regulatory Authority.

The development and updating of a Probabilistic Safety Analysis for each installation is in charge of personnel belonging to the plant, with the participation of specialists of the above mentioned section.

Such group has its own review procedures, and during certain stages of the studies, it also received help from independent reviews carried out by international organizations (such as the IPERS mission in 1996).

NASA also has engineering groups in the installations and in its headquarters, who are able to perform peer reviews of design modifications.

Concerning thermal hydraulic and neutronic analyses, NASA is assisted by CNEA's specialists who have participated in the elaboration or in the review of several projects. Moreover, it asks for international advice in subjects such as seismology, system reliability, etc.

Quality assurance programs of the installations are audited by quality assurance group depending on each installation's head. Besides, there is a central quality assurance group depending on the General Manager, in charge of NASA's audits.

N°: 25
CNS-Ref.-Art.: 10
Page of Report: -
Chapter of Nat. Report: -
Country: Spain

Is the requirement to give due priority to safety included in any legal document or in the operators policy documents? What are the ARN criteria to assess this matter?

The requirement related to priority to safety is implicitly expressed in the National Law of Nuclear Activity, as already mentioned in the National Report and in our presentation.

From the point of view of the Regulatory Body, priority to safety is clear enough taking into account that the first purpose since it was created was to protect people against harmful effects of ionising radiation and preserve radiological and nuclear safety in nuclear activities carried out in the country.

From the point of view of the Responsible Organization, priority to safety is evident in the contents of the document called Policies and Principles, issued as Annex 9 of our Report.

Additionally, safety priority can also be seen in the application of the regulatory control safety principles, the design safety principles and the operation safety principles that include the establishment of safety culture.

N°: 26
CNS-Ref.-Art.: 10
Page of Report: 4/6
Chapter of Nat. Report: 10.2.2., 10.2.5.
Country: Canada

The report states that the Regulatory Body and the Responsible Organization have adhered to safety culture principles.

This is implemented by management attitudes and policies.

Please describe the safety culture in the operating organization, and give few examples on how it is promoted by rules, procedures and management actions.

Commitment to safety culture of the operation organization is expressed in the document Policies and Principles of the company, approved by its Board of Directors in September 1997, which to this respect asserts that “it is an objective of the company to promote safety culture”.

Besides, an important effort on the subject is devoted to diffusion and training in all the organization stages during the last years. For example:

During 1997 and 1998, both in CNA I and CNE, training courses on safety culture were dictated, and most of the installations personnel attended them (some 7000 man-hours).

During 1998 a specialists meeting on safety culture and quality assurance took place in Embalse.

Since 1997 a systematisation the reports on recordable events was implemented. Such reports are prepared in the operation area and are then used in the training area.

Self assessments have been also carried out enabling the determination of the development of the organization in aspects of safety culture, such as: communications, group work, relationship manager-staff, relationship with third parties, attitude to face problems, conflict management, etc.

N°:27
CNS-Ref.-Art.:11
Page of Report:7
Chapter of Nat. Report:11.2.4., Table 11.5
Country: Canada

The report states that the financial resources needed by the Responsible Organization are provided from energy sales, and from the United Fund which is decided by the Energy Secretary. What are the bases for deciding the United Fund, and how are they related to the adequacy of funding for the Responsible Organization?

NASA is a company belonging to the State, which must act in the wholesale electric market.

On the other hand, it must contribute with an annual canon to the National Atomic Energy Commission.

The amount of the funds coming from the United Fund is decided taking into account such considerations when the yearly budget is elaborated. The amount of this fund is authorised by the Energy Secretary.

N°: 28
CNS-Ref.-Art.: 11
Page of Report: 8
Chapter of Nat. Report: 11.3
Country: Germany

What is the current state of negotiations of procurement of funds for backfitting measures, and what timetable is expected for realisation?

Funds for back-fitting come from credit organizations. Both such funds and the corresponding investments will be made effective in three stages.

The first stage has already been completed. Investments corresponding to the second stage are in progress.

Presently, negotiation is carried out to make the third stage effective.

N°: 29
CNS-Ref.-Art.: 11
Page of Report: 9-11
Chapter of Nat. Report: 11.7
Country: Canada

The report describes the NPP personnel qualification.

Please describe the access of the NPP personnel to advisory technical, engineering and scientific services if available.

As already mentioned in the Report, each nuclear installation has an engineering section covering most of its needs concerning technical support required for the plant operation.

On the other hand, the Responsible Organization also has an engineering service organization that satisfies some of the needs of technical support for nuclear installations.

In order to cover other significant engineering aspects that cannot be covered by any of the two mentioned groups, it is possible to contract engineering services from domestic or foreign companies.

For any other kind of support associated with specific research or development, nuclear installations and the Responsible Organization can rely on the help coming from National Universities and the National Atomic Energy Commission.

N° 30
CNS.Ref.-Art. 11
Page of Report: 10
Chapter of Nat.Report: 11.7.1
Country: Germany

Is it planned to install a full-scope simulator for CNA I

For the time being, no full-scope simulator installation is foreseen for Atucha I, being Nuclear Angra II in Mambucaba (Brazil) full-scope simulator used for that purpose.

A partial graphical-interactive simulator has been installed and is operating in Atucha I, in order to carry out specific personnel training in it.

N°: 31 + 32
CNS-Ref.-Art.: 12
Page of Report: 1-5/2
Chapter of Nat. Report: 12/12.2
Contry: Canada/Germany

***Canada:* The report focuses on the system and equipment aspects of the facility in the correction and prevention of human errors. Please explain how personnel related issues, such as shiftwork systems and work control methods, have been addressed by the Responsible Organization. How does the Responsible Organization determine if human factors issues need to be investigated in the incident analysis, and what are the related management activities?**

***Germany:* Are there working groups installed in the NPPs or within the operating organization to deal with Human Factors aspects?**

Up to date the analysis of incidents produced by human errors -for instance, operator's failures by using non adequate operation procedure- produces corrective actions that generate review or modification of operation procedures, hardware modification on man-machine interface, or retraining and qualification of operation personnel.

The analysis of the root causes, included within the operational experience program implemented by the end of 1998, sets a systematisation of the activities carried out nowadays. Such program includes an ad-hoc group of specialists for the investigation of human factors' contribution to an incident in areas of: procedures, training, oral communications, organizational factors, human engineering and supervision.

Besides, the objectives of a program of self assessment related to human behaviour is being discussed between the Regulatory Body and the Responsible Organization. Such program would be carried out by the Responsible Organization and the fields of interest are: plant organization, operation, maintenance and training.

N°: 33
CNS-Ref.-Art.: 13
Page of Report:
Chapter of Nat. Report: 13.1, 13.2.
Country: Japan

In the report of the Article 13, there is a comprehensive description about approaches of quality assurance. In the section 13.1, there is a description such that quality assurance programs are mandatory for the installation. In the section 13.2, it is stated that work was done on the adaptation of the new quality concepts and principles during 1996 and 1997.

On the other hand, a lot of persons and organizations are involved and multi-layered contractors exist for the construction and operation of the nuclear installations generally.

Under these circumstances, what kinds of efforts are exerted in order that all persons and organizations involved become thoroughly familiar with the requirements?

Qualification and training are the mostly used means to achieve the objective of making the personnel be familiar with new concepts and safety requirements.

In addition to diffusion of general principles of quality and safety culture, the involved staff is trained each time a new procedure is introduced or changes performed.

This practice, usual in the installations, is being intensified in support sections of the Responsible Organization.

The incorporation of quality culture is more difficult for temporary personnel. Such personnel receive specific qualification and training before starting their duties.

Besides, their performance is mainly evaluated when carrying out tasks related to programmed maintenance. Such evaluation is used the appointment of personnel is required for future tasks.

Besides, tasks carried out by temporary personnel are supervised by staff belonging to the operation organization, who must give example of understanding the requisites of the job to be done.

N°: 34
CNS-Ref.-Art.: 13
Page of Report: 3
Chapter of Nat. Report: 13.2, Table 13.1
Country: Canada

The report shows the status of the Operation Organization General Quality Assurance Program as of July 1988.

What is the schedule for the organizational units to complete the revisions planned? Are these revisions consistent with the ISO-9000 series?

Are computer codes used for safety analysis covered by verification and validation requirements?

First of all, as we have already stated it in the National Report, Table 13.1 is updated to July 1998 (not 1988).

The following table includes dates required in the first part of the question.

Organization Unit	Document	Revision	Numbre of Procedures
NASA	General Manual	Rev. 0 Updated 21/11/97	General Procedure 10
CNA I	Manual of Quality Assurance in Operation	Rev. 1 Under review Planned: December 1999	200
CNE	Manual of Quality Assurance in Operation	Rev. 3 Updates: 2/2/99	460
CNA II	Manual of Quality Assurance in Construction	Rev. 1 Under review	60
Engineering and Services	Manual of Quality Assurance of the Services Department	Rev. 4 Updated8 27/11/97	150
SEU Project	Manual of Quality Assurance for the SEU Project	Rev. 2 Updated: 1/12/98	8

These revisions are not consistent with ISO-9 000 standard but with the General Manual of Quality Assurance of the Responsible Organization, with the AR 3.6.1.- regulatory standard and with the 50-C-Q Code of IAEA.

In the particular case of computer codes used for safety analysis in Argentinian nuclear power plants, the situation is as follows:

The thermal hydraulic and neutronic codes used in the country are those widely used by the international community.

Most of them have been obtained as a consequence of Argentinian participation in the corresponding international projects.

In case the code is used in scenarios identical to those already used in other countries, no additional validation is considered necessary.

In case the code be used in particular scenarios, the set of input data modelling such particular case and its corresponding output are validated against plant transients observed for each particular case.

Such validations are then recorded and presented to the international community in the corresponding meetings.

Regarding code verification, this corresponds to the situation in which the source code is modified.

In case such modifications are needed, they are verified, in first place, by programer who is responsible for the change. In second place, and if it corresponds, the modification is presented in an international forum for its approval and incorporation in the new versions of the code.

After the verification is performed, validations of observed plant transients are carried out for each particular scenario, according to what was explained in the preceding paragraph.

N°: 35
CNS-REF.-ART.: 13
PAGE OF REPORT: 5
CHAPTER OF NAT. REPORT: 13.2.1.3.
COUNTRY: CANADA

The report states that the Responsible Organization promoted international reviews of the nuclear power reactor CNA I operation quality system.

Were reviews of the operational quality system performed at other nuclear installations? What were the main findings of the WANO peer review for CNA I, and how were they used?

Besides the OSART mission for Embalse Nuclear Power Plant in November 1997, a WANO peer review was carried out in CNA I in 1995.

As it is usual in peer reviews, positive aspects of the plant management are pointed out, and improvements in areas in which deficiencies were detected are proposed.

The most positive aspects mentioned are:

- ✓ Strong motivation and commitment of personnel in search of excellence.
- ✓ Good level of technical knowledge of the operation staff about plant systems and their interactions, maintenance and plant engineering.
- ✓ A predictive maintenance system with equipment and procedures that enable the prevention of components and system degradation
- ✓ An organization of technical support personnel that provided good results in the execution of plant tasks.
- ✓ Good materials and maintenance resulting in high plant availability and excellent working conditions.

Among the aspects that should be improved the following can be mentioned:

- ✓ Greater interaction with other installations, including those of different types, in operational experience items.
- ✓ Reinforcement of knowledge and techniques leading to a better use of the ALARA concept.
- ✓ More emphasis in making staff conscious of the principle of priority to safety.
- ✓ Increase of news reports to problems and events of minor importance or consequences, with the purpose of improving feedback of operational experience and detecting early precursors and negative trends.
- ✓ More diffusion of expectations and managerial policies among personnel in order to achieve a complete participation in reaching the goals.

These general comments and some others about specific aspects were take into account by the installation head, who implemented a set of actions, the positive results of which were recognised by the WANO follow up mission carried out in 1997.

N°: 36
CNS-Ref.-Art.: 13
Page of Report: 13-6
Chapter of Nat. Report: 13
Country: Brazil

Is there any effort to develop a QA program for the Regulatory Body, including internal audit procedures?

In fact, the implementation of quality practices in the organization of the Regulatory Body was gradually initiated some years ago.

Subjects such as procedure manuals have been developed for the regulatory activity, including related administrative and regulation aspects, inspectors qualification and training, records, document control, inner and outer interfaces and managerial self evaluation.

Regarding this last item, it should be mentioned the Regulatory Body has a Planning and Prospective Unit, depending directly on the President of the Board of Directors, mainly having the following functions :

- ✓ Analyse the degree of fulfilment of objectives and tasks programmed.
- ✓ Assess the human and financial resources use.
- ✓ Analyse the effectiveness of managerial processes and regulatory actions.

In addition, other unit, the Internal Auditing Unit, is responsible for the use of budget resources during the fulfilment of regulatory functions.

Both units are complementary.

The goal to achieve is to have an effective Quality System, preventing pathological deviations in quality assurance (for instance, a quality system based on a great number of reports that do not help in the regulatory effectiveness).

N°: 37
CNS-Ref.-Art.: 14
Page of Report: 1
Chapter of Nat. Report: 14.1
Country: Canada

The report states that the Responsible Organization performs different studies throughout the life of the plant to maintain an adequate safety level.

What are the studies performed to assess the effects of aging on the validity of analysis and the reliability of safety components?

What are the reporting requirements that cover safety analysis updates, radiological monitoring, periodic inspections, reliability, and R&D?

Availability control of safety systems and safety related systems are routinely carried out in Argentinian nuclear power plants on the basis of:

- ✓ Periodic tests.
- ✓ Inspections and predictive maintenance.
- ✓ Non-destructive tests.

Results are compared with the original goals established in reliability studies.

As regards ageing, for instance aspects related to erosion-corrosion, they are controlled by the assessment of rates of wall thickness decrease or loss of material, based on prospective concerning expected lifetime and comparison with allowed design threshold.

The eventual replacement of piping or other components are possible actions to be taken.

Results of periodic tests are used for PSA feedback, as well as to evaluate ageing effects in components, equipment and systems, for example containment isolation.

As regards ageing, the following items are mentioned, in addition to the RPV follow up:

- ✓ Replacement of neutron flux in-core detectors.
- ✓ Inspection and replacement of heat exchanger pipes belonging to the emergency cooling system.
- ✓ Follow up of the state of weldings in input and output nozzles of coolant and moderator (RPV).
- ✓ Follow up of the state of the steam generator.

Additionally, the implementation of the Ageing Management Program with the objective of providing systematic framework to predict and detect any degradation put in practice last year, will improve the ageing analysis related to safety of nuclear power plants.

On the other hand, the installation operation licences contain explicit requirements and requisites regarding the update of the final safety report and the PSA periodically.

Such re-evaluations help in the assessment of reliability analyses and data on components, equipment and systems.

N°: 38
CNS-Ref.-Art.: 14
Page of Report: 4
Chapter of Nat. Report:
Country: Brazil

What were the legal powers of TUV Baden during the inspection of CNA I commissioning.

Baden TUV was contracted by the National Atomic Energy Commission to verify the quality of components of German and Argentinian manufacture, and to certify progress in fabrication and construction of CNA I.

N°: 39
CNS-Ref.-Art.: 15
Page of Report: 1
Chapter of Nat. Report: 15
Country: Germany

What is the importance of a parameter: collective dose per generated energy?

Standard AR 3.1.2 establishes that enough retention of radioactive effluents shall be provided to ensure that the committed effective dose to the critical group due to discharge of such effluents shall not exceed 0.3 mSv per year, and that the collective dose shall not exceed 15 mSv man per MW year of electric energy generated.

Therefore, the importance of the use of dose limit per unit of electric energy generated is mainly regulatory.

The limitation is imposed not only to the dose to the critical group but also on collective dose in general. Besides, the parameter collective dose per unit of electric energy generated enables a comparison among collective doses of nuclear installations operating at different power levels, and it is used in some countries as a global measure of controlling effluent emission to the environment.

N°: 40
CNS-Ref.-Art.: 15
Page of Report: 3
Chapter of Nat. Report: 15.3
Country: Germany

What is the reason for using the value of 10 000 USD/manSv for planning optimisation of radioactive discharges?

The value of 10 000 USD/Sv-man is used in the differential cost-profit analysis for the optimisation of radiological protection systems. It should be added that the systems of radioactive effluents retention of nuclear installations must be optimised 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/Sv-man and 25 000 USD/Sv-man.

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.

N°: 41
CNS-Ref.-Art.: 16
Page of Report: 7
Chapter of Nat. Report: 16.5
Country: Canada

The reports mentions emergency situations due to external threat e.g. “warlike acts”. which measures are thought of in this particular case?

The reference to “warlike acts” found in the corresponding section is related to the organization in charge of managing any kind of significant catastrophe, name Civil Defence, which performs its function in a generic way and not especially for nuclear installations.

The paragraph pretends to briefly summarise the origin (by the end of the Second World War) and evolution of the Civil Defence system in the country.

On the other hand, Physical Protection standards clearly show that design of physical protection systems shall not consider direct measures to repel an assault of military or paramilitary groups, so that it is not considered necessary, from the point of view of physical protection, to adopt specific measures to face this particular case.

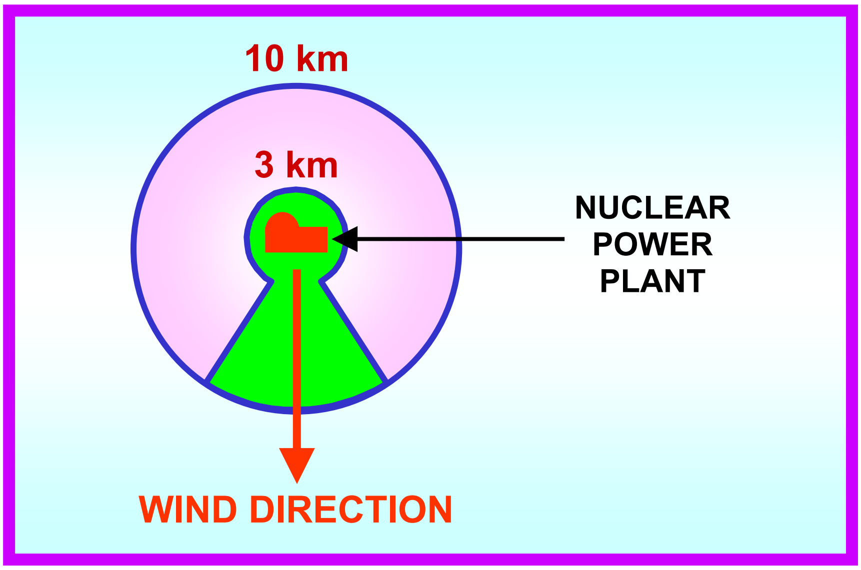
It should be also mentioned that since 1976, the Argentinian nuclear installations are guarded by a safety organization called National Gendarmerie (in charge of looking after the country's frontiers), which assigns personnel for each of the nuclear installations, taking care of their perimetric region.

N°: 42
CNS-Ref.- Art 16
Page of Report: 16-2
Chapter of Nat.Rep.: 16
Country:Brazil

What are the criteria used to start using iodine tablets during an emergency? (please confirm that this does not depend on radioactive measurements, as stated in 16.2)

The Regulatory Body considers two types of countermeasures in case of a nuclear accident:

Countermeasures of automatic application within an area of 3 km around the installation, and in a circular sector with angle 60° in the wind direction, centred at the installation.



- ✓ Non urgent measures, strongly dependent on the resulting measurements of radioactive substances dispersed to the environment.

The iodine prophylaxis constitutes a measure of automatic application and therefore it does not depend on radioactivity measurements.

Its application only depends on the meteorological conditions and of the installation state.

Stable iodine pills are distributed in the form of potassium iodine pills to people staying within the region previously identified as key hole.

The consideration of iodine prophylaxis as an urgent measure, without the requirement of previous radioactivity measurement, is based on the fact that their effectiveness is strongly dependent on the quickness of their ingestion.

A delay in the ingestion could lead to a decrease of its efficiency.

Although iodine ingestion could not be innocuous for a small group of people, when all the information available is analysed, it is concluded that the risk associated to its ingestion is negligible as compared to the one associated to the presence of iodine in the cloud, due to an accidental emission to the environment.

The sequence for the implementation of this urgent countermeasure is:

A. based on the information regarding the installation state and meteorological conditions, stable iodine pills are distributed
B. once the emission begins, and without waiting for field measurements or description of the source term, their ingestion is ordered.

N°: 43
CNS-Ref.-Art.: 16
Page of Report:
Chapter of Nat. Report: 16
Country: Spain

Evacuation is included among non-urgent emergency measures.

May this fact be interpreted as if no evacuation were required in the first phase of emergency? if so, what is the process followed to arrive to this conclusion?

Non massive evacuation criteria is adopted during the initial hours of an accident.

This decision is based on the fact that the protection level for the first hours (mainly to balance the consequences of the passage of the radioactive cloud) is ensured with urgent or automatic countermeasures of access control, sheltering and iodine prophylaxis application.

It should be mentioned that one of the non urgent protective measures is the compulsory evacuation of those zones affected by the deposition of radioactive material if the radiation level exceeds 100 mSv integrated over the 6 hours following the emission, being the evacuation optional if such value is not exceeded during the first 24 hours following the emission.

This measure tends to avoid the population from receiving high integrated doses due to radioactive deposit.

People could afterwards suffer deterministic effects if they are subject to not very high dose rates but during long periods of time. In order to decide evacuation measures for this case, field measurements are required.

N°: 44
CNS. Ref.-Art. 17
Page of Report: 1-9
Chapter of Nat. Report:
Country: Canada

The report decribes activities related to site evaluation.

Given that siting studies include evaluation of the NPP impact on the environment, please describe the related evaluation criteria and safety philosophies. Are there any international arrangements with neighbouring countries that affect site evaluation?

The pre-operational studies for the evaluation of the impact a nuclear power plant produces on the environment are oriented to determine the population distribution, the climatic conditions, the hydrologic and hydro-geologic conditions as well as the activities developed in the zone influenced by the installation.

This information leads to the determination of the most exposed group of people in the surroundings, dilution factors and the routes of transfer of radioactive materials to the environment during normal or eventually accidental operation of the installation.

As concerns normal operation, it enables the establishment of discharge limits, and with the actual values at hand, it is possible to determine dose to people as a consequence of operation.

Under accidental conditions, this information is necessary for the preparation of an off-site emergency plan and for the implementation of countermeasures needed to mitigate the accident consequences.

Concerning the second part of the question, it should be mentioned that there is an Agreement of mutual co-operation between Argentina and Brazil, including a program of immediate notification for the case of a nuclear accident.

Besides informal contacts are maintained and an agreement of general nature with the Oriental Republic of Uruguay has been signed. None of them affect the site assessment.

N°: 45
CNS.Ref.-Art. 17
Page of Report: 1-9
Chapter of Nat.Report: 17.6
Country: China

Why does Argentina re-assess the site of NPP?

Those characteristics of the site that could potentially affect safety may require a re-evaluation during the installation lifetime, if there is new information available or due to modification of criteria or methods initially used or resulting from advances in practices and the evolution of the state of the art concerning siting.

As a result of a re-evaluation, modifications in the surroundings may be known that require a updating of any of the safety related parameters of the installation, particularly if the occurrence of new external events could affect it.

As an example of the usefulness of such kind of re-evaluations, and concerning Atucha I, it may be mentioned that the installation of a new dam upstream the plant site, produced new flooding conditions that could cause an impact on the site.

Another example, concerning Embalse, was the site re-evaluation related to seismic, tornadoes and severe storms, because the data base containing meteorological information had been elaborated 20 years ago, and an updating was necessary to confirm that the design basis had not been modified.

N°: 46
CNS-Ref.-Art.:17
Page of Report:
Chapter of Nat. Report:
Country: Spain

Attention is given to maintaining the continued acceptability of the site regarding site parameters like meteorological, hydro-geological or seismic.

Is there a surveillance program for assessing the validity of other parameters like population and industrial uses ?

An updating of parameters related to population and soil uses in the region influenced by Atucha I and Embalse is in progress.

Such updating comprises the population distribution values, the agriculture -cattle breeding production and the development of other human activities, particularly in the surroundings of the plant (first 10 km), with resolution of 1 km^2 up to 25 km and of 100 km2 up to 100 km, and 500 km for agriculture- cattle breeding respectively.

The purpose of the study is to have updated parameters enabling the best effectiveness in the application of the Emergency Plan, a risk evaluation for the population in case of an eventual nuclear accident and improvement of the estimation of dose incurred by the critical group under normal operation of nuclear installations, as well as to evaluate the potential occurrence of man induced events, or generated by modifications to conditions existing when the installation was designed.

N°: 47
CNS-Ref.-Art.: 19
Page of Report: 3
Chapter of Nat. Report: 19.3.1
Country: Canada

The report states that the conditions for authorisation of the commercial operation of nuclear facilities were documented in the operating license. What is the regulatory process for operating licence renewal?

The operation licence for nuclear power plants is forever in force -so that no renewal is necessary- except if it is cancelled, suspended or modified by the Nuclear Regulatory Authority.

Its validity depends on the compliance with conditions established in the licence and in other requirements issued by the Regulatory Body. To this respect, one of the most important requirements establish that the Responsible Organizations must periodically review its Safety Report and its Probabilistic Safety Analysis.

The non compliance with the conditions or requirements submitted produces, as a consequence, the application of the corresponding sanctions. Once solved the non compliance causes, the licence can be issued again under the conditions imposed by the Regulatory Body.

N°: 48
CNS-Ref.-Art.: 19
Page of Report: 3
Chapter of Nat. Report: 19.3.1
Country: Canada

The report states that the Operating Manual defines the station-specific operational limits and conditions.

What are other station-developed response procedures such as alarm manual, impairment manual, and radiation protection manuals?

The mandatory documentation of Atucha I consists of a set of manuals, programs and other documents concerning the installation operation, being one of them the Operation Manual.

The possible causes of warning and alarm activation, the subsequent actions to be followed and the instruction for operation under abnormal conditions, are established in the two volumes constituting the above mentioned manual, named Manual of warnings and alarms, and Instruction for the operation of the plant under abnormal conditions.

Other mandatory documents are, the Code of Practices (constituted by the radiological protection and nuclear safety procedures), the Manual of Monitoring (in-site and environmental), the Quality Assurance, the Periodic tests program or the In-service inspection Program.

N°: 49
CNS-Ref.-Art.: 19
Page of Report: 4/7
Chapter of Nat. Report: 19.3.5., 19.4.5.
Country: Germany

Is there a complete list of all significant events available which occurred in CNA I and CNE, also according to INES-scale?

The INES scale is used in Argentina with the unique purpose of providing public information.

Such scale is not related with safety decisions making.

N°: 49 – CNA I Description of the event	Degree of importance (INES)	Date of ocurrence of the event
Unavailability of one of the branches of the boron injection	0	15/2/95
Failure in the temperature measurement of the axial bearing of one of the main cooling pumps.	0	24/3/95
Coolant pumps failure due to failure in the electric supply.	0	09/02/96
Main cooling pumps failure due to low seal flow.	0	15/02/96
Reactor scram due to fast turbine closing as a consequence of a failure in the voltage regulation system.	0	13/01/97
Scram due to alternator failure.	0	20/03/97
Main cooling pumps failure due to seal breakage.	0	03/03/98
Main cooling pumps failure due to low seal flow.	0	02/07/98

N°: 49 – CNE Description of the event	Degree of importance (INES)	Date of ocurrence of the event
High pressure of oil regulation in the turbine control system.	0	26/01/95
Loss of off-site 500 and 132 kV lines.	0	07/10/95
Pressure tube damages during garter spring relocation.	1	08/10/95
Out-of-service situation due to the trip of a pump of condensed gas extraction.	0	17/12/95
Pressure tube failure and small loss of coolant during start up after garter spring relocation.	2	18/12/95
Opening of main steam safety valves.	1	08/01/96
Steam generator tube failure.	0	30/06/96
Turbo-generator exciter blockage.	0	31/01/97
Loss of heavy water due to tubing failure.	0	29/05/98
Out-of-service situation due to vapour loss in steam generator purge valve failure.	0	03/10/98
Wrong measurement on level in the main condenser.	0	28/11/98

N°: 50
CNS-Ref.-Art.:19
Page of Report: 7
Chapter of Nat. Report:19.4.6
Country: Germany

For how long is spent fuel to be stored in the "silo field" and what future use is intended for the fuel?

Spent fuel storage in concrete silo fields is of temporary nature. For this reason, facilities for their withdrawal have been foreseen.

Silos life time has been estimated in a minimum of 50 years.

Unlike other intermediate storage methods, dry storage requires less operational supervision and preventive/corrective maintenance. Nevertheless, a program of periodic sampling is carried out in the inner silo atmosphere, in order to detect possible failures in the baskets containing fuel elements.

Meanwhile, the future use of spent fuel elements continues being evaluated.

Nº: 51
CNS-REF.-ART.: 19
PAGE OF REPORT: 7
CHAPTER OF NAT. REPORT: 19.4.6.
COUNTRY: CANADA

The report describes irradiated fuel and radioactive waste management. What means are used to minimize the generation of radioactive waste?

A way of minimising radioactive waste generation is by maintaining a low a fuel element damage rate, which means a smaller quantity of mechanical filters and ionic exchange resins used in cleaning the heat transport system.

The fuel element damage rate is kept low on the basis of the following actions.

- ✓ The manufacturer has a quality assurance program.
- ✓ Fuel elements are inspected by the manufacturer (self inspection) and by NASA inspectors (in-situ and periodically).
- ✓ When fuel elements arrive at the installation, all of them are controlled in order to detect eventual damages produced during their transport. Besides, their diameter is also checked.

In operation:

- ✓ Damaged fuel elements are immediately withdrawn from the core.
- ✓ Their presence is detected by means of a gaseous fission product sensor located in the coolant. The channel containing the damaged fuel element is identified using a delayed neutron sensor.
- ✓ Operation power is usually maintained in a stable regime, with power ramps when needed, according to design.

Personnel training in the procedure of radioactive waste management.

Compliance with segregation procedures of contaminated wastes.

Measurement, classification, segregation and compression of radioactive wastes.

Nº: 52
CNS-REF.-ART.: 19
PAGE OF REPORT: -
CHAPTER OF NAT. REPORT: 19.5.1.2
COUNTRY: CHINA

How is outside experience feedback in Argentina?

As a requirement of the Regulatory Body, the Responsible Organization developed a program of operational experience management in nuclear power plants, with the purpose of strengthen and impair such type of analysis taking advantage of domestic and international experience regarding nuclear power plant operation in a systematic, efficient and effective way.

The program contains:

- ✓ Maintenance of international information channels.
- ✓ Identification and report of abnormal situations that may have occurred in Argentinian installations.
- ✓ Establishment of criteria for selecting and evaluating information of abnormal events occurred abroad, that could be useful for Argentinian nuclear power plants.
- ✓ Follow up of corrective actions corresponding to abnormal events occurred.
- ✓ Internal and external communication of events requiring corrective actions.

Three operational groups take part in the organization carrying out such program: a central group (NASA) and two local groups, one at each operating nuclear power plant.

The central group is constituted by a co-ordinator, a technical secretary and five analysts (NASA); each local group is constituted by two engineers, a technical secretary, a shift head and an adequate number of analysts.

The central group systematises and categorises the information received according to its degree of application.

Events are priorised according to their influence on both the installation safety -related to the public and personnel- and its availability.

Besides, a second (and redundant) review is made in case the initially available information is considered non applicable.

Assessments identify deep causes of abnormal events, corrective actions to be taken and corresponding lessons learnt.

Information concerning abnormal events, corrective actions and their degree of implementation, etc., are updated in a data base.

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On the other hand, the organization carries out complementary tasks such as:

- ✓ Meetings between specialists of different areas at each plant and in other units.
- ✓ Courses and seminars on methods of analysing abnormal events.

- ✓ Information compilation about safety performance indicators in foreign plants and analysis of the corresponding safety trends, in order to provide information on the subject to local personnel.

N° 53
CNS.REF.-ART. 19
PAGE OF REPORT:
CHAPTER OF NAT.REPORT: 19
COUNTRY: SPAIN

Which is the ARN review and control process for design modifications being implemented by the operator?

Is there a permit before completion of the modification? Are there criteria to exempt the operator from this procedure?

Any significant component or procedure modification that could influence on the installation safety, on radiological protection, or could produce an deviation from the conditions established in the operation licence, must be previously authorised by the Regulatory Body prior to its execution.

To fulfil the requirement, each project of installation modification that could be safety significant, must be carried out according to specific procedures.

In fact, each project of this type is first evaluated by the Internal Advisory safety Committee of the plant. Documentation, technical and economical justification as well as an independent reviewer report are evaluated by the Technical Review Committee (independent of the licence holder).

If such evaluation is satisfactory, the project is submitted to the Regulatory Body for its consideration and approval.

When the design modifications proposed are complex, the different stages (conceptual design, detailed engineering, implementation, etc.) are submitted to the consideration of the Regulatory Body.

The Regulatory Body authorisation of each stage is necessary to proceed with the following stage.

Simple modifications or changes, to the project are also submitted to the consideration of ARN.

If ARN considers it necessary it appoints inspectors during the execution of the modification.

N°: 54
CNS-Ref.-Art.: 19
Page of Report: -
Chapter of Nat. Report: -
Country: Spain

Is there a systematic program, agreed with the ARN, for the management of ageing and plant life extension?

Until some years ago, operating nuclear power plants took into account ageing only in specific cases, so that such problem was considered in an isolated style and non systematically.

The Regulatory Body, understanding that a global treatment of ageing was necessary, required to the Responsible Organization -at the beginning of 1998- the implementation of a Program of ageing management with the purpose of predicting and detecting any degradation that could affect the safety margins taken into account in the design.

Such program considers the following aspects:

- ✓ Selection of safety related components for which ageing must be evaluated.
- ✓ Analysis of ageing mechanisms for components and/or development of practical methods for ageing monitoring.
- ✓ Evaluation of remaining life time for selected components and management of degradation due to ageing by means of practices of surveillance, maintenance and operation techniques, suggesting corrective or mitigation actions.

It should be said that the Responsible Organization presents the corresponding progress reports of the application of the Ageing Management Program, each three months.

NASA is presently under a privatisation process.

This is the reason why the decision regarding an eventual life extension for nuclear installations in operation will be analysed by the new licence holder.

Nevertheless, in order to obtain the authorisation for such extension from the Regulatory Authority, the Responsible Organization must demonstrate to the Regulatory Body that during the extension life time, safety levels (at least those originally established by design) are preserved, and, if corresponds, additional necessary requirements must be meet for the periodic update of such levels.

N°: 55
CNS-Ref.-Art.: 19
Page of Report: -
Chapter of Nat. Report: -
Country: Spain

It is stated that the procedures for accident management are presently being elaborated.

What is the schedule foreseen for its implementation.

The Responsible Organization has recently initiated the elaboration of procedures for accident management. Such elaboration comprises three stages:

- ✓ Conceptual design
- ✓ Basic design
- ✓ Development of instructions
- ✓ Implementation

During the conceptual design stage, solutions proposed by different countries concerning accident management are analysed, particularly the methodology applied for German nuclear power plants.

As regards basic design, results from probabilistic safety assessments for Argentinian plants, as well as characteristics of their systems, contribute to the choice of the most adequate measures for accident management.

In relation with development of instructions, this stage comprises the selection of a structure for the manual of accident management and the writing of the corresponding instructions.

Finally, although measures for accident management try to make use of every safety means in the plant, some back-fitting measures may be necessary.

At present, the first stage is in progress, and its end is estimated for mid 2000. Concerning the other two stages, only when the first one is completed it will be possible to establish their duration, due to the limited background, both domestic and international, available on the subject.

N°: 56
CNS-Ref.-Art.:
Page of Report: -
Chapter of Nat. Report: -
Country: Brazil

Is there any requirement related to periodical reassessment or re-licensing of nuclear power plant in the Argentina?

There are explicit requirements and requisites of the operation licence about the re-elaboration of the final safety report and the PSA re-evaluation for nuclear installations each five years. Such re-evaluations must contain plant modifications -both of systems and porcedures- and new data coming from their specific experience.

Besides, there are systematic programs considering subjects related to operational experience, ageing and assessment of the state of safety issues, specific for each plant.