República Argentina

NATIONAL NUCLEAR SAFETY REPORT

1998

CONVENTION ON NUCLEAR SAFETY

NATIONAL NUCLEAR SAFETY REPORT

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INTRODUCTION

I.1 GENERAL CONCEPTS

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 24776, 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 is carrying on since the beginning of its nuclear activities, 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 some 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 have recently been incorporated with slightly enriched uranium (0.85%), so the reactor core is partly loaded with slightly enriched uranium fuel elements and partly with natural uranium. The reactor is moderated and cooled with heavy water (see Annex 8).

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).

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 needs to be subject to national (or federal) jurisdiction and regulated as an organic and indivisible system. For this reason the National Congress is empowered to establish the laws concerning the subject, through Section 75 paragraphs 18 and 32 of the Constitution.

Within this context, Act No 24804, 1997 or "National Law of the Nuclear Activity", last year passed by the National Congress, is the legal framework for the peaceful uses of nuclear energy.

Article 1st of the Act No 24804, 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 organised, can be carried out both by the State and the private sector.

In the case of nuclear power generation, Article 4 of the Decree 1540, 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 NATIONAL PROGRAM CORRESPONDING TO NUCLEAR INSTALLATIONS

Safety assessments carried out by the Regulatory Body from the beginning of Atucha I and Embalse nuclear power plants operation, of which the most relevant results are detailed in this Report, indicate that no objection is found for continuing their commercial operation in compliance with the regulatory standards in force in the country, and usual international standards for the nuclear industry.

Those nuclear installations comply with the obligations established in the Convention on Nuclear Safety as may be concluded from this Report.

The Company in charge of the operation of the nuclear power plants is in a privatisation process through Act 24804, 1997; it foresees that the construction of Atucha II will be completed and its commercial operation initiated within a time term no longer than six years.

I.4 SUMMARY OF THE MAIN SUBJECTS CONTAINED IN THE REPORT

The present National Report has been performed in order to comply with Article 5 of the Convention on Nuclear Safety, and has been prepared as much as possible following the Guidelines Regarding National Reports Under The Convention on Nuclear Safety, approved in the Preparatory Meeting of the Contracting Parties, held in Vienna in April 1997.

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.

The information contained in the articles, which are part of the Report, and its complementary annexes show the compliance of the Argentine Republic, as a contracting party of such Convention, with the obligations assumed.

Article 6 describes the actions adopted by the organisation 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 or in response to regulatory requirements.

Article 7 presents the legal and regulatory structure that rules nuclear safety. It also analyses the criteria supporting the normative framework and the nuclear power plant licensing and regulatory control system.

Article 8 concerns functions, responsibilities and organisational structure of the Regulatory Body, human and economical resources, personnel qualification and their relationships with other official and private organisations.

Article 9 describes the Licensee's responsibilities and the regulatory controls required to verify the compliance with such responsibilities.

Article 10 analyses the policies and the priority to nuclear safety established by the Regulatory Body as well as by the organisation operating the nuclear power plants.

Article 11 deals with functions, responsibilities and the structure of the organisation in charge of the operation of the nuclear power plants, human and economical resources and personnel qualification.

Article 12 analyses the systems required to detect, prevent and correct human errors including the improvements to the man-machine interaction in the nuclear power plants control room.

Article 13 shows the quality assurance program in the design, construction and operation of nuclear power plants.

Article 14 deals with safety assessments performed by the organisation in charge of the nuclear power plants operation, maintenance activities and safety assessments and the verifications and safety related evaluations performed by the Regulatory Body at every stage of the nuclear installation life time.

Article 15 describes the basic radiological safety criteria used, the existing rules on the subject, the authorised discharge limits and dose evaluations to workers of nuclear power plants and to the public.

Article 16 describes the laws, regulations and requirements existing in the country and their implementation in case of a radiological emergency in a nuclear power plant. It analyses the actions to be taken inside and outside the plant, by all the intervening organisations, with special emphasis in training exercises on the emergency plan application.

Article 17 summarises the studies related to nuclear power plants siting and site re-evaluation studies

Article 18 analyses the existing standards concerning design and construction of nuclear power plants and their compliance with the principles of defence in depth, diversity and redundancy.

Finally, Article 19 analyses the mandatory documentation for nuclear power plants operation, the technical support given to the installations, the feedback mechanism of operational experience, fire protection and relevant events communication and irradiated fuel element management on site.

All the information contained in this Report has been updated to March 31st 1998 being no subsequent modifications incorporated except for those cases considered significant.

The terminology contained in this Report is, in general, consistent with that used in the International Atomic Energy Agency publications. The terms specifically related to the Argentine Republic regulatory system are the same as those used in its regulatory standards, as detailed in the corresponding glossaries (see Annex 1).

I.5 ANNEXES

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

Annex 1 has the complete set of the regulatory standards mentioned in the Report.

Annex 2 includes Act No 24804, 1997 or National Law of Nuclear Activity.

Annexes 3 and 4 include Atucha I and Embalse operating licenses, and Annex 5 includes Atucha II construction license.

Annex 6 includes IAEA Safety Review Mission Report at Atucha I nuclear power plant.

Annex 7 includes OSART mission report performed in CNE.

Annex 8 briefly describes technical characteristics of the nuclear power plants in operation in the country.

Annex 9 contains the Policies and Principles of the Company in charge of the operation of the nuclear power plants.

Annex 10 contains the list of technical documents used in the preparation of the present Report.

ARTICLE 6

EXISTING NUCLEAR INSTALLATIONS

6.1 INTRODUCTION

The country has two operating nuclear power plants, Atucha I (CNA-I) and Embalse (CNE), and another one under construction, Atucha II (CNA-II).

CNA-I, located some 100 km Northwest from Buenos Aires City, with a net electric power of 335 MW, began its commercial operation in 1974. The reactor is of the pressure vessel PHWR type. According to the original design CNA-I is fuelled with natural uranium, but fuel elements of a new design have recently been incorporated, having slightly enriched uranium (0.85 % w U-235), so that the reactor core is now partly loaded with slightly enriched fuel, and partly with natural uranium fuel. The reactor is moderated and cooled with heavy water (see Annex 8).

CNE, which initiated its commercial operation in 1984, is located in the Province of Córdoba, some 110 km to the south of the homonymous city, and its net electric power is 600 MW. It is a PHWR reactor of CANDU type, natural uranium loaded and heavy water moderated and cooled (see Annex 8).

Although the construction tasks have been interrupted, CNA-II has a net electric power of 693 MW, and is being built near CNA-I. It was also conceived as a PHWR type reactor, natural uranium fuelled and heavy water moderated and cooled.

In August 1994 the *Comisión Nacional de Energía Atómica* (National Atomic Energy Commission) was divided into three independent organisations: one of them retained the original name, National Atomic Energy Commission; it remains within the public sector and its current activities are related to research and development, fuel cycle, radioisotopes and radiation sources, and specialised training in nuclear subjects.

The second organisation named *Nucleoeléctrica Argentina S.A.* (NASA Company in charge of the operation of the nuclear power plants), is constituted by the branch of the former National Atomic Energy Commission which was in charge of nuclear power plant operation, and by an organisation named *Empresa Nuclear Argentina de Centrales Eléctricas* (ENACE Argentine Nuclear Company of Electrical Power Plants) acting as architect – engineer of CNA-II.

The third one, originally named *Ente Nacional Regulador Nuclear* (National Board of Nuclear Regulation) and afterwards *Autoridad Regulatoria Nuclear* (Nuclear

Regulatory Authority[1]) by means of the Act No 24804, 1997, is constituted by the regulatory branch of the former National Atomic Energy Commission. This branch started the regulatory activities in 1958. The Regulatory Body is a completely independent organisation, entrusted with all the regulatory functions (see Annex 2).

At that time (1994), requirements concerning improvements to radiological and nuclear safety in nuclear power plants in operation and under construction, issued by the regulatory branch of the former National Commission of Atomic Energy, were at different stages of fulfilment. Both the National Board of Nuclear Regulation and then the Nuclear Regulatory Authority sustained such requirements, thus preserving the institutional and legal continuity.

Those requirements were consequence of safety analyses, regulatory inspections and operational experience, mainly from the significant events occurred in domestic nuclear power plants as well as in other installations located abroad.

For instance, several safety analyses showed the need of adapting CNA-I emergency cooling system, designed during the late sixties, to the present nuclear safety criteria. This fact produced a series of requirements on the basis of an improving and updating (backfitting) program for the installation. Another example is related with the optimisation of radiological protection: several requirements were issued, particularly referred to CNA-I, where a program for the progressive replacement of all the fuel channels is implemented.

As an example of the operational experience gathered, the actions carried out as a consequence of the R06 fuel element channel breakage in CNA-I in 1988, and those carried out in CNE as a consequence of pressure tube breakage in the Canadian Pickering-2 nuclear power plant in 1983, may be mentioned. In the first case the Regulatory Body established special requirements previous to the operation restart; in the second case modifications to the specific program of inspection and follow-up of the reactor pressure tubes were carried out by the organisation in charge of the operation of the nuclear power plants.

These and some other improvements to the nuclear power plant safety, detailed in what follows, are at present the complete set of subjects included in the nuclear power plant safety examination existing at the time the Convention came in force.

[1] From now on, the name Regulatory Body will be used when referring to the Radiological and Nuclear Protection Regulatory affairs Management (see Section 7.2.1), the National Board of Nuclear Regulation or the Nuclear Regulatory Authority, but if necessary, these last names will be used.

6.2 SAFETY ASSESSMENTS AND CORRECTIVE ACTIONS IN NUCLEAR INSTALLATIONS

6.2.1 ATUCHA I NUCLEAR POWER PLANT

Some of the significant requirements established by the Regulatory Body are related to:

- Fuel element channel breakage.
- Enforcement of the AR 10.1.1 regulatory standard requirements.
- Coolant channel replacement.
- Second heat sink.
- Emergency power supply system modification.
- Plant specific probabilistic safety assessment.
- Pressure vessel integrity.
- Safety assessment due to the introduction of slightly enriched fuel elements.

6.2.1.1 Fuel element channel breakage

Due to the deformation or breakage of a water level measuring probe guide tube, in August 15th 1988, a coolant channel (R06 position) was broken, producing, as a consequence, a breakage of the fuel element located inside it and several damages on other internal reactor components. Nuclear safety was not affected due to the intrinsic safety characteristics of this nuclear power plant (negative reactivity coefficient due to moderator temperature), but from the availability point of view this event produced the nuclear power plant shutdown, until its total repair. The damages produced inside the moderator tank were limited to a few cooling channels, instrumentation probes, thermal isolating material and increase in the heavy water contamination. The plant stayed out of service till mid 1990 (that is during 16 months), and demanded a significant investment in specially designed and built equipment in order to face such repair.

The Regulatory Body decided to interrupt the nuclear power plant operation license validity until the internal damaged component were repaired and sensors of different types for early detection of similar events were installed. Furthermore, the Regulatory Body required the performance of the corresponding safety analysis, as well as a series of special tests to verify the operating conditions of safety and control systems.

In December 1990, once the test results were analysed, the Regulatory Body issued a new operating license containing, in addition, the conditions under which the nuclear power plant operation could be restarted at full power and a special monitoring program to enable the early detection of events similar to that occurred in August 1988.

The conditions established in the new operating license were the following:

- Periodic inspections of moderator tank, coolant channels, guide tubes for probes and lower distribution toroid of the moderator tank.
- Implementation of a program to enable early detection and alert of abnormal events related to eventual failures of fuel element channels or other internal components of the reactor tank, as well as possible failures in the cooling of fuel elements located inside such channels.
- Enhancement of the in-service inspection program, aiming at detecting eventual damages in material properties (e.g.: ageing, fragility, fatigue, formation or worsening of defects) and to the implementation of the required corrective actions.
- Additional periodic tests and special inspections, besides those already taken into account in the nuclear power plant maintenance program, in order to analyse the behaviour of components related to the above mentioned incident.

Once the repairing and test tasks were finished, and as a request of the Argentine Government, IAEA carried out an independent safety review mission in 1990 (Safety Review Mission at Atucha-I Nuclear Power Plant) in order to evaluate the nuclear power plant situation (see Annex 6). The experts group sent by IAEA reviewed the repairing tasks of the moderator tank internal components and evaluated monitoring techniques used during the repair as well as implications of the event on the installation safety. As a result that group issued recommendations taken into account in subsequent regulatory requirements.

6.2.1.2 **Enforcement of the AR 10.1.1 regulatory standard requirements**

The criteria contained in the new version of Standard AR 10.1.1 set in force by the Regulatory Body in December 1994, are in agreement with the ICRP recommendations contained in its Publication 60. Based on this standard, the Regulatory Body performed new radiological safety revisions in the operating nuclear power plants and decided to make some additional requirements to adapt them to the new criteria.

Nevertheless, some occupational doses in CNA-I cannot be substantially decreased until the channel replacement mentioned in Section 6.2.1.3 is completed. Meanwhile, it is estimated that due to their nature, the tasks related to moderator heat exchanger cleaning and maintenance will involve doses close to the limits, for the staff participating in their execution.

6.2.1.3 **Coolant channel replacement**

As in other nuclear power plants of the same generation, there are in CNA-I some reactor internal, primary and moderator circuit components submitted to several corrosion and erosion phenomena. In order to reduce their effects, the above mentioned components were covered with a non ferrous alloy containing 60% cobalt, commercially known as Stellite-6 (e.g. coolant channel guides and primary circuit pump and valve seats). The channel surface covered with this alloy is approximately 16 m^2, constituting around 95% of the total Stellite-6 present in the nuclear power plant primary circuit. As a result of the material activation produced by the core neutronic flux, cobalt-60 is formed.

On the other hand, corrosion and erosion products, most of which contain cobalt-60, are transported and deposited along the moderator and primary circuit.

Therefore, gamma emission due to cobalt-60 additionally contributes to the radiation field in certain places of the reactor building and consequently, to the increase of occupational doses (see Section 15.5.1.1). As a result of this, CNA-I normalised occupational dose is higher than in those nuclear power plants where Stellite-6 is not used.

The Basic Radiological Safety Standard AR.10.1.1 issued by the Regulatory Body, requires the preservation of dose limits and constraints, as well as the application of radiological protection optimisation. Thus, the Responsible Organisation (see Sections 9.1 and 9.2) was required to eliminate the cobalt-60 generation by totally replacing the original coolant channels by others without Stellite-6. This task is gradually going on at each programmed outage and up to now 130 channels have been replaced, representing more than 50 % of their total number, estimating that the job will be finished by the end of 1999.

6.2.1.4 **Second heat sink**

Accidents with a small loss of coolant (small LOCA) located in the cold branch of the primary circuit, had not been evaluated in detail in the original plant design. This type of accident was not taken into account due to the fact that CNA-I was designed according to the "Maximum Credible Accident" criterion, which only takes into account the maximum LOCA (Section 2F), according to the current criteria at that time. During a small LOCA, the primary circuit depressurizing is very slow, thus delaying the corresponding injection of water coming from the core emergency cooling system. This situation could be complicated by the presence of some debris elements that might still be present in the moderator circuit as a result of the event already described in 6.2.1.1.

These two factors and the need of increasing safety in terms of core damage probability determined in the level 1 probabilistic safety assessment, indicated the convenience of installing an additional core residual heat removal system, independent from the present one, through the secondary system via steam generators.

This improvement consists of providing the steam generators with an independent water supply system capable of removing the residual core heat during several hours, with the purpose of producing the necessary cooling gradient to manage different accidental situations.

The Regulatory Body has established the following minimum requirements to be fulfilled by such system:

A. To replace the moderator in its core cooling function, during shutdown.
B. To remove the residual core heat under the following plant conditions:
   a) Loss of normal and emergency power supply.
   b) Pump house unavailability.
   c) Intermediate residual core heat removal system unavailability.

Taking these requirements into account, the Responsible Organisation proposed the following characteristics for the new system design:

A. Two independent water supply trains to the steam generators and two lines of hot steam release to the atmosphere each having 100% capability, shall be provided.
B. The injection water temperature shall be room temperature, so that the deposit tank may be made of concrete.
C. The second heat sink pumps shall be impelled by diesel engines. Their power shall be enough to impulse electric generators, each of them being able to satisfy the complete system demand.

The Regulatory Body considered that the proposal was adequate, so that the Responsible Organisation decided to implement it. At present the basic engineering has already been developed and the detailed engineering of the second heat sink including controlled release to the atmosphere is being developed. It has been estimated that the modifications will be completed by 2000.

6.2.1.5 **Emergency power supply system modification**

The plant has three diesel generators of 1.6 MW each. In case of loss of normal power supply the generators operating with a 2 out of 3 logic, provide the energy demands of those systems necessary to remove the core residual heat. The engines start up requires an approximate time of 40 seconds. During this period the electrical power needed for the core heat removal, which cannot be interrupted due to a design condition, is supplied by a hydraulic turbine operated by the condenser feedback flow.

The Regulatory Body required the Responsible Organisation to improve the emergency power supply system reliability significantly, in order to reduce the annual probability of core damage in case of a loss of off-site power event, which is one of the most significant conceivable accidental sequences.

In response to this requirement, the Responsible Organisation implemented the electrical interconnection between CNA-I and CNA-II through their normal power busses. The Regulatory Body accepted such solution because as a result of its implementation, a considerable increase in the system reliability was achieved, limiting its use until the beginning of CNA-II commissioning tasks.

Such interconnection, which operates either in automatic or manual mode, incorporates a back up emergency power system and can restore the electric supply to CNA-I normal busses in a time of some 100 seconds. It also complies with the criteria of physical and functional independence and it is composed of two redundant trains, each one capable of satisfying 100% in demand.

6.2.1.6 **Plant specific probabilistic safety assessment**

The operating license issued when the repairing tasks related to the R06 channel failure (see Section 6.2.1.1) and nuclear power plant commissioning were finished, required, in addition, the performance of a level 1 probabilistic safety assessment, according to IAEA-50-SS-P4 guide.

The Responsible Organisation appointed a group of specialists, constituted by 12 professionals who started working in September 1992. The plant specific level I probabilistic safety assessment, which comprised the analysis of the internal plant events occurred during full power operation as initial condition (determination of the core damage probability), was finished in March 1996 and the updated version issued in December 1997 is now being reviewed.

In April 1997, an International Peer Review Service Mission (IPERS) was carried out by IAEA. All the IPERS recommendations were taken into account by the Responsible Organisation.

It should be mentioned that the probabilistic safety assessment enabled the identification of the main contributors to risk, and at the same time, it was possible to improve and update operational procedures and to improve the surveillance program.

The results of the probability safety assessment demonstrated some weaknesses in the plant design and operation. Even though such weaknesses were not of enough importance as to interrupt the plant operating license, they produced the issue of requirements by the Regulatory Body in order to perform corrective actions of immediate implementation. Such requirements have already been fulfilled by the Responsible Organisation. As an example, it should be mentioned the electrical interconnection between CNA-I and CNA-II (see Section 6.2.1.5), and the installation of isolation valves in the pressurizer spray lines to reduce substantially the frequency of occurrence of primary low pressure events. There are other requirements issued by the Regulatory Body of mid-term implementation like the construction of a second heat sink which have already started to be implemented.

The following are other improvements derived from the probabilistic safety assessment results:

- Improvements in the safety related river water cooling.
- Updating of emergency operation procedures, with the corresponding operator training.
- Improvements on reliability of the main feedwater system.
- Improvements in the shutdown core cooling system.

This probabilistic safety assessment, -that includes loss of coolant events classified in 5 groups, 17 plant transients and other initiating events for instance the pressure vessel failure-, made it possible to select the set of accidental sequences mainly contributing to core damage.

Since CNA-I has been operating for more than twenty years, the Regulatory Body had also issued a number of requirements related to the nuclear power plant backfitting and updating plan, aiming at increasing the safety level. The probabilistic safety assessment results showed that the fulfilment of such requirements contributed effectively to the increase of the safety level.

6.2.1.7 **Pressure vessel integrity**

CNA-I reactor pressure vessel surveillance program was initiated in 1974, by introducing 30 test specimens distributed in the lower reflector, inside the moderator tank. In 1980, 10 additional capsules containing samples of the same type, but made of A508 class 3 material[2] were introduced, aiming at obtaining a relative reference for neutron exposure. On the other hand, irradiated capsules were withdrawn and examined. No definite conclusion could be obtained about the pressure vessel material behaviour, due to the differences between the neutron spectra present on samples located inside the moderator tank and those spectra on the pressure vessel wall.

Due to the uncertainties associated with the previous results, Siemens - Kraftwerk Union AG carried out an irradiation program in the German VAK reactor in 1985. The experiments were carried out with probes made of the same pressure vessel

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[2] Forgin steel. ASTM A-508 Class 3 Ring Forgin -20 Mn Mo Ni 55 ASME SA508, cl3)

basis material as well as with A508 class 3 material, simulating the conditions on the CNA-I pressure vessel wall.

Although there are some uncertainties due to the short irradiation time (high acceleration factor), irradiation carried out by Siemens - Kraftwerk Union AG indicated that 35 °C was the temperature for ductile-to-brittle transition at the end of CNA-I lifetime (32 full power years). On the other hand, preliminary results of the most unfavourable LOCA analysis showed that the pressure vessel would always be at a temperature higher than 35 °C at the end of its lifetime.

The Regulatory Body has carried out an assessment of the available information related to the reactor pressure vessel. It was concluded that there are uncertainties regarding the reactor pressure vessel integrity under certain accidental operational situations.

On the other hand, NASA has initiated an evaluation of those accidental scenarios having the most unfavourable stresses for the pressure vessel integrity (Pressurised Thermal Shock analysis), as well as a program of the necessary actions (i.e. improvement in the surveillance program), to be carried out in order to minimise such uncertainties contained in the studies.

In addition, it is important to emphasise that the design characteristics of the plant do not facilitate an optimal mixture of the injection of the emergency system water, due to the fact that the loop seal always contains cold water, which has a negative contribution to the effects of a thermal shock during the eventual occurrence of an accidental situation.

It is the opinion of the Regulatory Body that, according to the present state of the art on the subject, it will not be possible to reduce the before mentioned uncertainties within the time period and in the way required. Therefore, with the purpose of ensuring that the reactor pressure vessel will continue preserving the appropriate safety margin, it has been required that the necessary measures should be taken in order to reach the year 2001 having heated the water contained in the high pressure accumulators of the core emergency cooling system.

6.2.1.8 **Safety assessment due to the introduction of slightly enriched fuel elements**

Originally, CNA-I standard fuel element is constituted by 36 natural uranium dioxide fuel rods, disposed in concentric rings having 1, 6, 12 and 17 fuel rods each, plus an additional structural rod located in the external ring.

The Responsible Organisation proposed, as a result of several technical and economic analyses, a replacement of the standard fuel element by another one having similar characteristics but slightly enriched uranium dioxide as fuel (0.85% w in U-235).

The advantage of using such slightly enriched fuel elements is a better use of the fuel through the higher achievable burn up, and consequently a smaller use of the refuelling machine, and a smaller number of irradiated fuel elements to be stored.

Nevertheless, the greater contents of the isotope U-235 (that is to say more fissionable material) imply certain differences with the standard fuel element, such as variations in global parameters, reactivity coefficients and kinetic parameters, having different response in operational events and transients. This is the reason why additional studies were necessary. The feasibility and safety analyses carried out by the Responsible Organisation showed that it was possible to add slightly enriched fuel elements in successive stages, each of them supported by the corresponding safety assessment. The stages are:

1.- During the first stage of the project, 12 slightly enriched fuel elements were loaded (representing 5% of the total number of fuel elements in the core), and it was shown that it was possible to operate the plant producing almost no changes in the essential safety related parameters. The fuel management strategy with three burn up zones (characteristic of the natural uranium fuel management) was preserved during this phase.

2.- The second stage included up to 60 slightly enriched fuel elements (25% of the total number approximately). The Responsible Organisation demonstrated that even with this number of slightly enriched fuel elements, it was possible to operate the plant under the same safety conditions it had been designed.

3.- Third stage included up to 99 slightly enriched fuel elements, that is some 40% of the total number of fuel elements.

There are now being carried out safety assessments, both by the Regulatory Body and the Responsible Organisation, with the purpose of including more than 100 slightly enriched fuel elements in the core.

6.2.2 EMBALSE NUCLEAR POWER PLANT

The Regulatory Body established requirements related to the following subjects:

- In-service inspection program.
- Pressure tube inspection.
- Enforcement of the AR 10.1.1 regulatory standard requirements.
- Dry storage for irradiated fuel elements.
- CALIN 122/84 Document.
- Plant specific probabilistic safety assessment.

6.2.2.1 **In-service inspection program**

CNE was designed taking into account normal and abnormal operational conditions, determined by knowledge and operational experience (in other nuclear power plants) existing when the project started. After its commissioning, the convenience of establishing complementary requirements for in-service inspections resulted from the acquired experience.

During operation, situations or operational events decreasing the lifetime of components and/or systems may eventually occur, which imply the necessity of introducing additional requirements in order to preserve the initial design conditions.

Such situations on components and/or systems may produce variations in thermal parameters, mechanical tensions, erosion phenomena, corrosion, irradiation, creep, hydrogen absorption, vibrations, friction, etc.; moreover, they may produce their premature ageing with the subsequent plant life time shortening and, under certain circumstances, if the proper corrective actions are not implemented, risk and unavailability of the plant may be increased.

Considerations of this type led the Regulatory Body to issue requirements to the Responsible Organisation, in order to review the in-service inspection program, to examine and verify the state of nuclear power plant components and/or systems, to enable the prediction of any eventual damage and determine the corresponding corrective actions to preserve safety functions.

The designer, Atomic Energy of Canada Limited, elaborated a program for the nuclear area based on the Canadian CSA No 285.4-M 1978 standard, named "Periodic Inspection Program Document" (18-PIPD), which takes into account ASME code, Section XI - Div. 1 and IAEA No 50-SG-02 guide recommendations.

On the other hand, and with the purpose using them at CNE, the contents of Canadian CAN3 N285.4 M-83 and CAN3 N285.4-94 Standards were incorporated to the program, including a fatigue factor for the establishment of inspection categories and the extent of its scope to steam generators, pressure tubes and feeders.

Concerning the conventional area (thermal cycle and auxiliary systems), the in-service inspection program is based on the recommendations of designers, Italian suppliers, Società Italiani Impianti P.A. Ansaldo and ANSI B31.1 standard.

Simultaneously to the development of such program, the Responsible Organisation carried out inspections related to operational events occurred during the nuclear power plant operation, also taking into account the operational experience in similar nuclear power plants.

In general, the in-service inspection program requires the inspection of the following systems:

(a) Coolant pressure boundary and systems the failure of which may produce unacceptable quantities of radioactive material release to the environment.

(b) Safety systems which are essential for reactor shutdown and fuel element cooling in case of a process system failure.

(c) Systems and components that may affect the integrity of the coolant pressure boundary or safety systems mentioned in (b).

6.2.2.2 **Pressure tube inspection**

Nowadays, CNE pressure tube inspection program takes into account, among other aspects, failures occurred at the Canadian nuclear power plants Pickering 2 in 1983 and Bruce 2 in 1986, as well as several events in which small coolant leakage were produced in other units of the same type. The Canadian companies Ontario Hydro and Atomic Energy of Canada Limited developed a wide research program with the purpose of determining the causes of such failures.

Problems requiring program elaboration as well as the corrective and preventive actions recommended by the mentioned entities were put into practice by CNE, and may be summarised as follows:

The material initially used for manufacturing CANDU reactor pressure tubes was a zirconium alloy named "Zircalloy-2". This material, used in other reactors, was discontinued when it was demonstrated that it was not suitable for long service periods, due to the growing corrosion and hydrogen absorption rates presented after some years of operation. It was then replaced by "Zr-2.5%Nb", another zirconium alloy without the "Zircalloy-2" problems, and having better mechanical properties; this material is presently used for manufacturing CNE pressure tubes.

When the pressure tube is in service, the material changes its properties and composition, due to both neutron fluence as well as coolant temperature and composition. One of the most significant changes is its increment of deuterium contents (or equivalent hydrogen). The equivalent hydrogen contained in the pressure tube zirconium may mainly produce two effects:

(a) Reduce the material ductility and tenacity.

(b) Favour the delayed hydride cracking when the material stress exceeds a threshold value.

Both effects happen only when the hydrogen equivalent concentration exceeds a limit value appearing as a solid phase hydride, a phenomenon that depends on the material temperature. Since solubility is a function of temperature and it is very low at room temperature, a certain solid phase hydride fraction is always present at low temperatures. However, solid phase hydrides will not appear at the channel operation temperature, while the equivalent hydrogen concentration is not sufficiently increased by the absorption of deuterium present at the primary coolant.

During manufacturing, the pressure tube material incorporates hydrogen in concentrations between 5 and 20 ppm. Under operation the equivalent hydrogen concentration is increased during the reactor lifetime, due to the corrosion phenomenon in the pressure tube internal side. The corrosion process releases deuterium from which approximately 5 % diffuse into the zirconium and the rest is dissolved and carried away by the heavy water circulating as coolant within the channel. Due to the presence of deuterium in the annular gas, a negligible quantity of it may also be incorporated to the pressure tube through its outer side.

Being the primary circuit pressurised and hot, the equivalent hydrogen contained in the pressure tube requires average concentration values near 55 ppm to precipitate as hydride (temperature at channel middle point of approximately 287 °C). But in the case of an eventual contact between pressure tube and calandria tube, the temperature at such point falls to 50 °C. Under such conditions, the equivalent hydrogen concentration required to initiate the blister formation process is around 25 ppm.

Thus, the three most important factors which determine the pressure tube sensitivity for blister development and growth are: equivalent hydrogen initial contents, deuterium absorption rate during operation and occurrence of cold zones, i.e. eventual contact between pressure and calandria tubes.

The equivalent hydrogen initial contents in the pressure tube could be estimated from measurements of samples obtained from the tube far end during its assembling in the reactor. The deuterium absorption rate during operation was determined by measuring it in already replaced tubes, or from samples of material obtained directly from the pressure tube inner surface by means of scraping techniques.

On the other hand, during the pressure tube assembling in CANDU 600 type reactors (CNE included), some garter springs shifted from their design position. These garter springs separate pressure tube from calandria tube, avoiding contact between them, enabling at the same time the annular gas displacement; moreover, the pressure tube axial shift is enabled due to thermal dilatation and creep effect. The mentioned displacement was probably produced by pressure tube oscillations induced by coolant and moderator circulation, before the channels were loaded with fuel, during the nuclear power plant cold commissioning.

Up to now there is no evidence that elastic rings may have shifted from their design position during the operation, after loading the channels with fuel.

The some garter spring shifts are of such magnitude that they allow contact between pressure and calandria tubes; this does not represent a problem itself unless the equivalent hydrogen concentration in the tubes exceeds the above mentioned limit value. In such case, the hydrides may gradually concentrate in the contact zone and eventually a blister will begin growing until the pressure tube breakage occurs due to working stress effects.

From the above considerations it is concluded that it is very important to eliminate contacts between pressure and calandria tubes, before the absorption deuterium process could produce blister development and the corresponding channel breakage. A special program was elaborated by Atomic Energy of Canada Limited. The Responsible Organisation created the Technical Committee of Channels Subprogram for CNE to co-ordinate the task related to the pressure tube inspection program.

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 CNE since the late eighties. Today this program is under execution, having been inspected 149 channels out of 380, estimating that the job will be finished by the end of 2003.

The Regulatory Body took notice of the events occurred in the Canadian nuclear power plants related to the pressure tube failures, particularly taking into account the results of the research program performed by Atomic Energy of Canada Limited and Ontario Hydro. At the same time, it was in close contact with Technical Committee of Channel Subprogram specialists, evaluating, from the safety point of view, the research results and consequences of the recommendations of such programs (see Section 19.5.1.2).

Particularly, a detailed follow-up of the procedure modification of humidity detection in the annular gas, implemented by CNE after Pickering 2 accident, was carried out as an effective means for the early detection of coolant losses ("leak before break" criterion). Furthermore, the Regulatory Body resident inspectors routine follow up the pressure tube inspections carried out during the programmed nuclear power plant outages aiming at relocating spacers in those channels requiring it.

During the programmed CNE outage of October - November 1995, an incident of interest related with the garter spring relocation occurred: due to electric failures in one of the coils of the repositioning tool named "SLARETTE Mark III", a cavity and a passing hole were produced at the inner side of A-14 and L-12 channels respectively. In the case of L-12 channel a coolant loss toward the annular space also occurred. Both pressure tubes had to be replaced. Due to the fact that the characteristics of this incident are directly applicable to all CANDU reactors, the Regulatory Body informed other regulatory authorities and IAEA Incident Reporting System.

6.2.2.3 **Enforcement of the AR 10.1.1 regulatory standard requirements**

The dosimetric history of CNE workers shows that only occasionally some worker received a dose exceeding 100 mSv in five consecutive years. Even in 1995, when an 8 weeks programmed outage took place, with 2 additional weeks added later, the highest recorded doses were about 20 mSv. Thus it seems that there will be no difficulties in the future to comply with the new occupational dose limits set by AR 10.1.1 standard.

6.2.2.4 **Dry storage of irradiated fuel elements**

CNE fuel elements are constituted by 37 "Zircalloy" tubes containing natural uranium oxide. The uranium dioxide (UO_2) fuel is constituted by little cylinders or pellets of 1.22 cm diameter and 1.60 cm long. The fuel element length is approximately 50 cm and it has 21.5 kg of UO_2 (19 kg of uranium). There are a total of 4560 fuel elements in the core, located inside 380 coolant channels. Each coolant channel has 12 fuel elements. The fuel element reloading is made during reactor operation.

The spent fuel elements are temporarily kept under water, inside pools located at the installation and capable of storing fuel elements during 10 years of nuclear power plant full power operation. Disassembling of adjusting bars, activity measuring and loading in transport containers of cobalt-60 (obtained by neutron irradiation of adjusting bars) withdrawn from the reactor core, are carried out in another pool.

After the minimum decay period established in six years, irradiated fuel elements are transferred to special dry storage silos, also located inside the nuclear power plant site. The fuel elements are introduced in stainless steel baskets, each of them containing up to 60 fuel elements vertically arranged in a circular grid; this operation is carried out under water. Later on, the baskets are sent to the transfer building where the lid is weld. Finally they are introduced in a special container providing enough shield and containment (transfer "flask") to be transported to the silo field where they are stored. Each silo contains 9 baskets.

The regulatory activities related to the Dry Storage of Irradiated Fuel Elements System comprised the design, commissioning and operation stages. During the design stage requirements on several aspects were made, specially those related to the system thermal parameters. The pre-operational cold and hot tests were authorised by the Regulatory Body and controlled by inspectors of the same organisation. Moreover, requirements concerning supervision and operation of the system staff qualification were carried out.

As a Regulatory Body request, an evaluation of the radiological consequences of possible accidents, particularly the eventual "flask" falling down during fuel element transportation to the silo was carried out. Even though such event has a very low occurrence probability, its analysis made it possible to incorporate new preventive actions to the on-site emergency plan.

Before the beginning of the Dry Storage of Irradiated Fuel Elements System operation, the Regulatory Body modified the nuclear power plant operating license, including additional conditions related to operation, physical protection and safeguards. On the other hand, resident inspectors control Dry Storage of Irradiated Fuel Elements System operation as a part of the inspection program.

6.2.2.5 **CALIN 122/84 Document**

The Regulatory Body performed a regulatory control during the CNE construction and commissioning stages. Controls included in situ follow-up of the significant activities related to civil engineering activities, evaluation of safety related systems, components and equipment, regulatory audits, seismic verification of structures and safety related systems, consultations and discussions of common interest subjects with the Canadian Regulatory Body, and even some operators of similar Canadian nuclear power plants.

At the time the operating license was issued, in January 1984, the Regulatory Body had, as far as it was possible, a complete knowledge of the installation safety design characteristics and operation. However, some aspects were at that moment being evaluate while others were still awaiting for a response from the Responsible Organisation, none of them being a condition for the issued of the above mentioned license.

A regulatory document issued with all these aspects, through which the Regulatory Body emitted, in 1984, a series of particular operation requirements including different subjects to be analysed by the nuclear power plant Responsible Organisation. Some of them are here mentioned:

- Responsible Organisation analyses about subjects concerning nuclear power plant safety.
- Necessity of performing safety evaluations or enhancing the existing ones, particularly concerning the reactor auxiliary systems.
- Evaluation of the benefits of incorporating trip parameters in the safety systems in addition to those foreseen by design.
- Information concerning operational experience in similar nuclear power plants related to operation systems and procedures.
- Analysis of the event occurred in June 1983, during commissioning, which affected the steam generator water supply system.
- Seismic verification of channels with coupled refuelling machines.
- Analysis and plan of action derived from the event occurred in Pickering 2 nuclear power plant, in August 2nd, 1983.

In October 1990 a complete review of the compliance with requirements included in the document was carried out, and subsequently some partial reviews were performed. At present most of the significant requirements have been fulfilled. Nevertheless, the probabilistic safety assessment results being carried out at CNE (see Section 6.2.2.6) will enable the fulfilment of the pending items in CALIN 122/84 Document.

6.2.2.6 **Plant specific probabilistic safety assessment**

The Regulatory Body required the Responsible Organisation the elaboration of a Level 1 probabilistic safety assessment, aiming at evaluating CNE safety level, in order to identify those areas requiring improvements, compare the safety level with national and international levels and help in the nuclear power plant operation.

Such study comprises two phases. The first one considers a full power reactor and its core as a radioactive release source. The second phase considers a subcritical state (shutdown) reactor including the analysis of fuel element (dry and wet) storage, irradiated fuel elements and cobalt bar transfer and radioactive waste storage.

The first phase is in an advanced state of elaboration, estimating that it will be finished by the middle of 1999.

6.2.3 **ATUCHA II NUCLEAR POWER PLANT**

CNA-II was designed by the Siemens - Kraftwerk Union AG, with the participation of ENACE Argentine Nuclear Company of Electrical Power Plants as architect engineer at the time the project began.

CNA-II construction license was issued in July 14, 1981. At first the construction was carried out normally, but from mid 1986 to the beginning of 1993 the evolution of civil engineering activities was very slow and consequently, activities concerning the licensing process were reduced. Nevertheless, the interaction between the Regulatory Body and the Responsible Organisation has been constantly active. Particularly, the experience obtained during CNA-I operation, which was transferred to CNA-II project, was a subject of continuous regulatory interest, reflected in the following actions:

- Replacement of Stellite-6 from the CNA-II reactor internal components and from its primary and moderator circuit (taking into account the experience acquired in CNA-I nuclear power plant).
- Modifications to the fuel element channel design and other internal components of CNA-II reactor, derived form the experience gained after CNA-I R06 channel failure.

The regulatory activities performed up to now are listed as follows:

- Preliminary safety report assessment.
- Preliminary risk analysis assessment.
- Quality assurance audits performed to the Responsible Organisation, to ENACE Argentine Nuclear Company of Electrical Power Plant and to certain contractors.
- Regulatory inspections performed in situ during manufacturing and installation of large components.
- Regulatory inspections performed to civil engineering activities and to component storage.
- Evaluation of mandatory documentation.

CNA-II construction has been interrupted and its completion and commissioning are related to the mentioned privatisation process (see Section I.3).

6.3 ACTIONS LEADING TO SAFETY IMPROVEMENT

6.3.1 CONTINUOUS EXECUTION ACTIVITIES

The continuous execution activities are the same for both operating nuclear power plants and comprise:

- Documentation updating.
- Organisation updating.
- Component inspection program.
- Periodic tests program.
- Emergency plan (see Section 16.1).
- Training and qualification of operating personnel, (see Sections 11.7, 11.7.1 and 11.7.2).
- Quality Assurance Program (see Section 13.1).

6.3.1.1 Documentation updating

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

6.3.1.2 **Organisation updating**

The nuclear power plant organisation evolves adapting to the successive stages each plant goes through, aiming at improving the response to each step requirements.

In the case of CNA-I, the necessity of updating and improving the nuclear power plant led to some changes in the organisation among which the following are mentioned:

- Constitution of a robotics group, for special maintenance tasks in areas of difficult access and with high radiation fields taking into account ALARA principle.
- Constitution of a probabilistic safety assessment group (see Section 6.2.1.6.).
- Enhancement of a plant engineering section to implement the backfitting (see Sections 6.2.1.4. y 6.2.1. 5).
- Transformation of the re-training and qualification area aiming at fulfilling the increasing demands of staff training and competence (see Sections 11.7, 11.7.1 and 11.7.2).
- Improvement of the Quality Assurance Sector.

The original organisation at CNE has not required so many changes. Nevertheless, some changes were made such as enhancement of the plant engineering section and the constitution of an operation group for the Dry Storage of Irradiated Fuel Elements System. In the first case, the organisation improvement enabled the updating of operation and maintenance procedures taking into account the experience acquired at the simulator.

6.3.1.3 **Components inspection activities**

These activities include the follow up of the reactor components, with the purpose of detecting eventual modifications of material properties, such as ageing, erosion, fragility, fatigue and defect formation. These modifications may be originated by the following processes: stress, thermal cycles, temperature, radiation, hydrogen absorption, corrosion, vibrations and friction.

The activities comprise every system and component of the nuclear and conventional area, which, according to applicable standards and to operating experience, are considered critical for the installation safety and availability. The applicable standards and codes are:

- Nuclear power plant operating license.
- Section XI, Division 1 of the ASME code.
- In- Service Inspection of Nuclear Power Plants, 50-P-2 of IAEA.

The activities are mainly carried out during both nuclear power plant programmed outages and during shutdown periods lasting several days before start up. Up to now no abnormal situations have been noticed.

6.3.1.4 Periodic tests program

Both nuclear power plants have a periodic tests plan as part of the surveillance program with the help of which both availability of safety and accident mitigation systems are periodically controlled.

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

6.3.2 INSTALLATION IMPROVEMENTS

Apart from the improvements corresponding to the continuous execution activities before mentioned, several modifications are carried out, contributing to safety improvement. Some of them are mentioned in what follows:

In Atucha I Nuclear Power Plant

(a) The installation of isolation valves in the auxiliary pressurizer spray lines.

(b) Improvements to the Electrical System.
- Location of barriers against fire between diesel generator groups.
- Location of collecting pots for flammable liquids.
- Replacement of grid lids for blind lids for wire protection.
- Replacement of control room doors by doors capable of resisting fire during 120 minutes.
- Closing of the communication between diesel room and hydrogen treatment plant by means of new solid bodies.
- Barriers to delay fire propagation in the medium tension bar (6.6 kV) room and in the battery room.

Additional improvements will be implemented when the fire risk analysis is finished (see Section 19.6.1).

In Embalse Nuclear Power Plant

(a) Replacement of 153 neutronic flux sensor assemblies by others of new design.

(b) Modification of dividing plates of the steam generator bottom dome plates.

Besides, specific safety issues are been analysed such as:

Hydrogen behaviour containment, void reactivity in loss of coolant accidents, fire protection (included in CNE probabilistic safety assessment program - see Section 19.6.1) and pressure tube failure with loss of moderator.

6.4 OPINION OF THE REGULATORY BODY CONCERNING THE OPERATION CONTINUITY OF NUCLEAR INSTALLATIONS

CNA-I and CNE comply with the regulatory standards related to design and operation; for this reason they have obtained their corresponding operating license. In the case of CNA-I, the Regulatory Body conditioned the operating license validity to the performance of some improvements detailed before, presently being carried out.

6.5 FINANCIAL ASPECTS

The above mentioned improvements to safety require financial resources to obtain supplies and services, be them either national or foreign. The sources of such resources are the following:

- Own funds coming from the electrical generation sale.
- Credits issued by a group of banking entities.

These latter are assigned to the obtention of supplies and services to be used in the improvement of CNA-I, specially in the updating and backfitting program.

6.6 COMPLIANCE WITH THE OBLIGATIONS IMPOSED BY THE CONVENTION

From the preceding considerations, it becomes evident that since the beginning of nuclear activities in the country, periodic and detailed safety assessments and improvements are carried out in the nuclear power plants. Therefore, the country complies with the obligations imposed in Chapter 6 of the Convention on Nuclear Safety.

ARTICLE 7

LEGISLATIVE AND REGULATORY FRAMEWORK

7.1 NATIONAL LEGISLATIVE FRAMEWORK

The National Atomic Energy Commission was created by Decree No 10936, 1950, one of its specific functions being to control the official and private atomic research carried out in the country.

Later on, different legal rules determined National Atomic Energy Commission's competence as regulatory body in the field of radiological and nuclear safety, particularly in those aspects concerning the individual and environmental protection against the harmful effects of ionising radiation, the nuclear installations safety and the control of nuclear materials use. The main legal rules concerning these aspects are Decree-Law No 22498, 1956, ratified by Act No 14467, 1958, and Decree No 842, 1958.

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

On the other hand, Decree No 842, 1958, approves and puts in force the Regulations for Using Radioisotopes and Ionising Radiation, with the purpose of regulating the use and application of radioactive materials and the radiation emitted by them or by reactions and nuclear transmutations, making clear that National Atomic Energy Commission will control the application of these regulations and will sanction the cases of violation. The use of x-ray generators was excluded from the National Atomic Energy Commission's competence, belonging exclusively to the area of the Ministry of Health

As a consequence of the increasing development of nuclear activity in the country, the functional independence of the Regulatory Body from National Atomic Energy Commission's other activities became stronger.

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

Based on such considerations, the Executive Power, supported by Act No 23696, 1989 and by Section 99 Paragraph 1 of the Constitution, created the National Board of Nuclear Regulation by Decree No 1540, 1994.

From this decree on, the National Board of Nuclear Regulation was the regulatory authority in radiological and nuclear safety, safeguards and physical protection along the whole national territory, performing all the regulatory and control functions of the nuclear activity, which formerly belonged to the competence of the National Atomic Energy Commission's regulatory branch.

Act No 24804, 1997 (National Law of Nuclear Activity) (see Annex 2) was passed, proclaiming (in its Article 7) the creation of the Nuclear Regulatory Authority. This authority is in charge of nuclear activity regulations and control concerning radiological and nuclear safety, safeguards and physical protection, giving, in addition, advice to the Executive Power on subjects of its competence.

The Nuclear Regulatory Authority, as an autarchic entity within the jurisdiction of the Presidency of the Nation, has full legal power to act in the fields of public and private rights, being the successor of the regulatory branch of the National Atomic Energy Commission and the National Board of Nuclear Regulation.

7.2 NORMATIVE FRAMEWORK

7.2.1 INTRODUCTION

Act No 24804, 1997 empowers the Regulatory Body to issue and establish the standards, which regulate and control nuclear activities, of compulsory application along the whole national territory.

Initially the attribute to dictate standards were assigned to the National Atomic Energy Commission, national organisation which created for that purpose the Radiological and Nuclear Protection Regulatory Affairs Management Branch. By Decree No 1540, 1994 the National Board of Nuclear Regulation was created. This board incorporated the personnel belonging to the above mentioned Branch and acted as an autarchic organisation until Law No 24804, 1997 was passed, creating the Nuclear Regulatory Authority with all the attributes and purposes prescribed by such Law.

The first regulatory standards related to nuclear power plant licensing were initially produced some twenty years ago and were known as "CALIN standards" (acronym for *Consejo Asesor para el Licenciamiento de Instalaciones Nucleares* - Advisor Council for the Licensing of Nuclear Installations -, a body belonging to the regulatory branch of National Atomic Energy Commission in that period). In the course of time a normative system was established comprising subjects such as radiological and nuclear safety, safeguards of nuclear materials and physical protection. The system, known as "AR Standards" (AR meaning "*Autoridad Regulatoria*" - Regulatory Authority), has at present 51 standards among which 28 concern (direct or indirectly) to nuclear power plant licensing in each of its stages:

design, construction, commissioning, operation and decommissioning (see Annex 1). The codes and names of the before mentioned 28 standards are shown in Table 7.1.

Table 7.1 - AR Standards concerning nuclear power plant licensing

AR Code	Name
0.0.1	Licensing of Major Installations
0.11.1	Conditions to Obtain Specific Authorisations for Personnel of Major Installations
0.11.2	Requirements of Psychophysical Aptitude for Specific Authorisations
3.1.1	Occupational Exposure in Nuclear Power Plants
3.1.2	Limitation of Radioactive Effluents
3.1.3	Radiological Criteria Related to Accidents in Nuclear Power Plants
3.2.1	General Safety Criteria in the Design
3.2.3	Fire Protection
3.3.1	Reactor Core
3.3.2	Heat Removal Systems
3.3.3	Pressure Primary Circuit
3.3.4	Fuel Element Behaviour in the Reactor
3.4.1	Protection System and Safety Related Instrumentation
3.4.2	Shutdown Systems
3.4.3	Confinement Systems
3.5.1	Emergency Electric Power Supply
3.6.1	Quality System
3.7.1	Documentation to be Presented to the Regulatory Body until the Commercial Operation of a Nuclear Power Plant
3.8.1	Pre-nuclear Commissioning
3.8.2	Nuclear Commissioning
3.9.1	General Safety Criteria in Operation
3.9.2	Communication of Significant Events
3.10.1	Protection against Earthquakes
3.17.1	Nuclear Power Plant Decommissioning
10.1.1	Basic Radiological Safety Standard
10.13.1	Basic Standard of Physical Protection of Materials and Nuclear Installations
10.14.1	Guarantee of Non Deviation of Nuclear Materials and Materials, Installations and Equipment of Nuclear Interest
10.16.1	Transport of Radioactive Materials

7.2.2 BASIC CONCEPTS

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

The risk information sources used by the Regulatory Body derive from reliability studies, probabilistic safety analysis and risk-informed operational insights. The plant-specific probabilistic safety analysis is the most important tool from the point of view of safety, since it ensures an effective regulation and a well-balanced inspection practice. As it was mentioned, the regulatory system is established using the performance-based approach.

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

7.2.2.1 **Deterministic and probabilistic aspects of the regulatory standards**

The installation's radiological and nuclear safety is conceivable by means of two approaches: one deterministic and the other probabilistic. The deterministic approach considers that an installation is safe when its design, construction and operation are able to face any of the events of a set of postulated accidental events, assuming the impossibility of occurrence of unforeseen accidents.

According to the probabilistic approach any type of accident may occur with certain probability being even considered the occurrence of an accidental situation not foreseen, as opposed to the deterministic approach.

Both approaches are complementary and the modern trend on radiological and nuclear safety is to use them together in a balanced manner. In this sense the Regulatory Body is leader in using both approaches at international level, having adopted, two decades ago, the probabilistic criterion of risk acceptance while maintaining some deterministic requirements.

7.2.2.2 **Basic criteria of radiological and nuclear safety**

The basic criteria in which radiological and nuclear safety is supported are being applied since long time ago and they are coherent with the ICRP recommendations (in its publications No 26 and No 60) (see Section 15.1).

On the other hand the Regulatory Body has contributed to formulate recommendations issued by international bodies (such as IAEA and ICRP), so that it is usual to find, in its own standards, concepts dealing with radiological and nuclear safety that appear in such recommendations.

7.2.2.3 **Radiological emergencies**

In case of emergencies the Regulatory Body applies criteria consistent with ICRP recommendations formulated in its publications No 60 and No 63 (see Section 16.2).

7.2.2.4 **Responsibility for safety**

The regulatory system considers that the organisation known as Responsible Organisation, is fully responsible for the radiological and nuclear safety of the installation. The mere compliance with the regulatory standards does not exempt the organisation from the mentioned responsibility. For this reason the regulatory standards are not prescriptive but, on the contrary, they are "performance-based" standards, that is to say, they establish the fulfilment of safety objectives; the way of reaching these objectives is based on engineering experience, on the qualification of designers, constructors and operators and on suitable decisions taken by the Responsible Organisation itself. Therefore the Responsible Organisation must demonstrate and convince the Regulatory Body that the installation is safe (see Sections 7.3 and 9.2).

7.3 LICENSING SYSTEM

7.3.1 GENERAL ASPECTS

A basic aspect of the regulatory system is the approach adopted, in which the Responsible Organisation deals with the design, construction, commissioning, operation and decommissioning stages of the nuclear power plant, being completely responsible for radiological and nuclear safety of the installation as well as for physical protection and safeguards.

The regulatory standards establish that the construction, commissioning, operation or decommissioning of a nuclear power plant shall not be initiated without the corresponding license, previously required by the Responsible Organisation and issued by the Regulatory Body. The validity of such licenses is subordinated to the compliance with the stipulated conditions included in the corresponding license, and to the standards and requirements issued by the Regulatory Body. The non-compliance with any of these standards, conditions or requirements may be enough reason for the Regulatory Body to suspend or cancel the corresponding license validity, according to the sanction regime in force (see Section 7.4).

The nuclear power plant personnel must be adequately trained and qualified according to their functions in the installation. The Regulatory Body requires also that the personnel assigned to safety related tasks be licensed.

7.3.2 LICENSING PROCESS

7.3.2.1 **Nuclear power plants licensing**

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

The construction license is issued when standards and requirements of siting, basic design and expected safety operation conditions have been complied prior to the beginning of this stage. It is a document through which the Regulatory Body authorises the nuclear power plant construction under the established conditions, and shall be fulfilled by the Responsible Organisation.

The applicable standards, consistent with international recommendations on the subject, establish the safety criteria to be observed in the design of the installation and define the timetable and type of mandatory documentation that shall be presented together with the application for a construction license (Standard AR 3.7.1). In particular the nuclear power plants design must comply with the radiological criteria related to accidents (Standard AR 3.1.3).

Once the construction license is requested by the Responsible Organisation, a continuous interaction between the constructor or operator of the future installation and the Regulatory Body is initiated. It is an iterative process, as complex as the demands involved. It should be emphasised that the Responsible Organisation's capability to carry out its responsibilities is evaluated since the construction stage.

Before Act No 24804, 1997 was passed, the Regulatory Body authorised the commissioning by issuing a specific authorisation once the corresponding requirements were fulfilled. The above mentioned Law establishes that the Regulatory Body shall issue a commissioning license.

The commissioning license establishes the conditions for fuel and moderator loading, operation with increasing power up to its nominal value, as well as verifications and tests of the components, equipment and systems to determine whether they comply with the original design basis. To do so the Responsible Organisation must appoint an *ad hoc* Commissioning Committee constituted by acknowledged specialists, who will continuously evaluate the execution of the commissioning program and recommend its prosecution if applicable (Standards AR 3.7.1, AR 3.8.1 and AR 3.8.2).

The operating license is issued when the Regulatory Body verifies that conditions, standards and specific requirements applicable to a particular installation are fulfilled. Such conclusion will be the result of analysing the submitted documentation and detailed studies, inspection reports carried out during construction and commissioning and the ad hoc Commissioning Committee recommendations.

The operating license is a document by which the Regulatory Body authorises the commercial operation of a nuclear installation under stipulated conditions, which shall be fulfilled by the Responsible Organisation (Standard AR 3.9.1). The non fulfilment of any of the imposed requirements without the corresponding Regulatory Body authorisation would imply the application of sanctions, that could lead to the operating license suspension or cancellation (see Section 7.4).

At the end of its lifetime and under the Responsible Organisation's request, the Regulatory Body authorises the ending of the nuclear power plant commercial operation and issues a decommissioning license. In this document, conditions for the nuclear power plant safe dismantling are established, being the Responsible Organisation in charge of planning and providing the necessary means for their fulfilment (Standard AR 3.17.1).

The evaluations performed prior to issuing a nuclear installation license include mainly aspects of quality assurance, construction procedures, operation procedures, previsions for in-service inspections, etc. Besides, emergency plans shall be prepared in co-ordination with corresponding National, Provincial and Municipal bodies (see Sections 16.1 and 16.5).

7.3.2.2 **Nuclear power plants personnel licensing**

AR 0.11.1 and AR 0.11.2 standards set the criteria and procedures to provide individual licenses and specific authorisations to the personnel who will apply for licensable functions in nuclear installations. Besides, both standards establish terms and conditions according to which the Regulatory Body may issue such individual licenses and specific authorisations, after the analysis and corresponding report of its advisory committees.

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

- *Individual License*: it is a certificate of permanent nature recognising the technical-scientific qualification necessary for a person to perform a certain function within the operation chart of a certain type of nuclear installation. The individual license is a necessary but not a sufficient condition for occupying a licensable position in a given nuclear power plant.

- *Specific Authorisation*: it qualifies a licensed person to perform such function in a particular nuclear installation. It has a maximum validity of two years and may be renewed.

Whenever an individual license or a specific authorisation is needed for its personnel, the Responsible Organisation shall submit the necessary documentation to the Regulatory Body requiring it. The "*Consejo Asesor para el Licenciamiento del Personal de Instalaciones Relevantes*" (CALPIR - Advisory Committee for the Licensing of Major Installation Personnel -), which advises the Board of Directors of the Regulatory Body concerning these matters, evaluates each applicant's qualification, and either suggests the issue of the requested certificate or produces a requirement to the Responsible Organisation otherwise, for the applicant's training and achievement of the needed qualification.

The persons who apply for an individual license or a specific authorisation or for the renewal of this latter must fulfil a number of requisites concerning qualification, working experience, training, re-training and psychophysical aptitude, which will depend on the installation and on the function. These requisites may be summarised as follows:

TO OBTAIN AN INDIVIDUAL LICENSE, IT IS REQUIRED:

Basic qualification: an education level (secondary, tertiary or post-graduated) suitable to enable the access to higher stages of qualification according to the technical scientific capability required considering the type of task and function level.

Specialised qualification: the technical-scientific knowledge in the nuclear field required to perform a licensable function adequately. The specialised qualification must respond to programs accepted by the Regulatory Body and to the approval of examinations with the participation of personnel of such body.

Working experience: significant experience for the correct performance of the function applied for.

TO OBTAIN OR RENEW A SPECIFIC AUTHORISATION, IT IS REQUIRED:

A suitable license for the function

Specific qualification: knowledge regarding radiological safety, the installation operation and characteristics, responsibilities of the position to be licensed and the mandatory documentation. The extension and depth of the applicant's knowledge shall be such that contributes to the safe operation of the installation. The specific qualification may be obtained after taking courses according to programs accepted by the Regulatory Body and approving examinations with the participation of personnel of such body.

On-the-job-training: to have carried out tasks corresponding to the function applied for, under the supervision of licensed personnel, in the same or a similar installation.

Re-training: to take courses and periodic evaluations prepared by licensed personnel in the operation of a nuclear installation, with the purpose of updating knowledge and developing aptitudes which will enable the applicant to face eventual abnormal situations.

Psychophysical aptitudes: the applicant's psychophysical conditions shall be compatible with the psychophysical profile needed to perform a licensable function correctly.

7.3.3 REGULATORY INSPECTIONS AND AUDITS

From the early days of the nuclear activities in the country, the Regulatory Body has performed assessments as well as multiple and different regulatory inspections and audits as frequently as considered necessary, with the purpose of verifying that nuclear installations satisfy the standards, licenses and requirements in force.

Act No 24804, 1997 "National Law of Nuclear Activity", authorises the Regulatory Body to continue with such inspections and regulatory assessments, performed by its personnel as:

- *Routine inspections* are essentially carried out by resident inspectors and other Regulatory Body inspectors. Their purpose is to verify that the Responsible Organisation complies with limits and conditions of operation established in the operating license (see Section 14.2.1.1).
- *Special inspections* are carried out by Regulatory Body's specialists in several subjects (dosimetry, instrumentation and control, thermohydraulics, etc) in coordination with resident inspectors. These inspections are performed under special circumstances or due to the occurrence of certain events in the installation. They have several purposes, e.g. to control preventive maintenance tasks during a nuclear power plant programmed shutdown (see Section 14.2.1.2).
- *Safety Assessments* are performed by Regulatory Body personnel and consist of the analysis of data obtained during an inspection or any other sources. For instance, radiological safety evaluations carried out during certain practices at the nuclear power plant, in order to detect their eventual weak aspects and identify possible ways of reducing doses to personnel (see Section 14.2.1.4.).
- *Regulatory Audits* are performed according to written procedures and are programmed in order to analyse organisation, operation and process aspects related to radiological and nuclear safety (see Section 14.2.1.3).

7.3.3.1 INSPECTIONS FOCUSED ON PROBABILISTIC SAFETY ASSESSMENT

Routine and special inspections focus on probabilistic safety assessment results so that special attention can be drawn to plant weaknesses or to the main contributors to risk. An important effort in routine inspections is the monitoring of parameters directly related to safety (particularly, the follow-up of the surveillance program applied to safety related systems and components), while special inspections focus on design modifications implemented with the purpose of improving safety levels at the nuclear power plant (the selection of which results from the plant risk assessment). Inspections are prioritised according to plant-specific probabilistic safety assessment results, taking into account qualitative as well as quantitative aspects.

The regulatory inspection team must have a consistent basis to determine which issues are important for the plant safety, considering the combined effects of design features and operational practices, the likely occurrence of each event and its consequences. In the case of plant backfitting, it is necessary to evaluate the relative significance of each issue and to determine their improvement priorities.

Resident inspectors at the nuclear power plants have a basis, which includes the study of the probabilistic safety assessment techniques. Safety assessment results, comprising aspects associated with risk insights, are discussed with resident inspectors to establish regulatory follow-up actions and also for training purposes. In some cases, the issues resulting from probabilistic safety assessment and having a significant impact on safety are incorporated into the inspection procedures followed by resident inspectors

7.3.4 REGULATORY ACTIONS

The regulatory actions that the Regulatory Body may take in relation with a particular installation are originated in three main causes:

- The results of regulatory assessments and inspections carried out in the installation.
- The knowledge of abnormal events occurred in the installation itself or in a similar installation.
- The application of the technical evaluation of the Regulatory Body itself.

In such cases, the Regulatory Body sends a regulatory document to the Responsible Organisation, which takes the form of a requirement, a recommendation or an additional information request according to the case; in it the Responsible Organisation is urged to carry out the required corrective actions in a certain time period. Such documents have the following scope:

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

As it has been already mentioned, the validity of a license is subject to the compliance with stipulated conditions, standards and other documents issued by the Regulatory Body; if any of these requirements were not fulfilled, this would be enough reason for the Regulatory Body to cancel or interrupt the validity of the corresponding license.

7.4 SANCTIONS REGIME

Act No 24804,1977 enables the Regulatory Body to apply sanctions and to suspend or cancel the validity of construction, commissioning, operating and decommissioning licenses, in the case of non fulfilment of standards, licenses or any other regulatory requisites.

The sanctions, according to the severity, are increasingly ordered as follows: warning, penalty to be applied in proportion to the seriousness of the fault, and in relation with the potential or real harm, suspension of the license or individual authorisation or even their cancellation.

According to the above mentioned Act, the Regulatory Body has put in force a sanctions regime to be applicable during construction, commissioning, and decommissioning of nuclear power plants.

The Regulatory Body appeals to the consensus and conviction when regulatory demands are posed to the Responsible Organisation. It is the philosophy of the Regulatory Body to consider that a sanction is not a routine regulatory action but the ultimate measure to be adopted at a conflictive situation.

7.5 COMPLIANCE WITH THE OBLIGATIONS IMPOSED BY THE CONVENTION

In the country a legislative framework has been established and maintained to govern the nuclear installations safety. Such framework provides:

- An appropriate set of regulatory standards to be applied in safety subjects.
- A licensing system.
- An assessment and inspection system to verify compliance with standards and requirements.
- A sanction regime to be applied in case of non-compliance with licenses, standards or other requirements.

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

ARTICLE 8

REGULATORY BODY

8.1 FUNCTIONS AND COMPETENCE OF THE REGULATORY BODY

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

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

Act No 24804, 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. It also establishes that the Nuclear Regulatory Authority has autarchy and complete legal capability to act in the field of private and public rights, and that its resources are basically integrated with regulatory fees and with State support.

Article 16 of the mentioned act establishes the functions and responsibilities of the Regulatory Body, authorising it to issue regulatory standards concerning radiological and nuclear safety, physical protection and safeguards. This article also assigns the Regulatory Body a series of functions, referred to radioactive and nuclear installations, already mentioned in Section 7.1 of this report.

On the other hand, Article 16 of Act No 24804, 1997 establishes a consulting procedure with the Responsible Organisation of nuclear installations, every time new regulatory standards are proposed or already existing ones are modified.

Act No 24804, 1997 gives the necessary legal competence to the Regulatory Body to establish, develop and apply regulatory standards to every nuclear activity carried out in the country. In order to guarantee a proper control level, such legal competence is complemented with a suitable technical capability. Namely, the

Regulatory Body has the capability to independently evaluate by itself the construction, commissioning, operation and decommissioning of nuclear power plants.

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

Besides, the Regulatory Body is empowered to contract specialists who can advice about subjects related to the specific fulfilment of its functions.

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

- Issue of the corresponding standards.
- Execution of regulatory inspections and audits to verify the compliance with granted licenses and authorisations.
- Independent execution of analyses and studies for the licensing process of nuclear installations.
- Development of technical and scientific aspects associated to radiological and nuclear safety.
- Training of personnel involved in radiological and nuclear safety, either belonging to the Regulatory Body or those working in installations, which perform practices under regulatory control.

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

8.2 REGULATORY BODY ORGANISATIONAL STRUCTURE AND HUMAN RESOURCES

According to the provisions in Articles 17 and 18 of Act No 24804, 1997 the Regulatory Body is managed and administrated by a Board of Directors constituted by six members, one of them being the President. All members of the Board are appointed by the Executive Power, two of them under proposal of the Deputies and the Senators Chamber respectively, and with recognised technical and professional background on the subject. The members of the Board have full time dedication and their appointment lasts six years, being a third of them renewed every two years.

The Regulatory Body acts like an autarchic organisation under jurisdiction of the Presidency of the Nation.

The Regulatory Body organisation is matrix based, where the different tasks involving different groups are designed as projects or activities, for a better use of the available economic and human resources. Activities are permanent works along years (i.e. regulatory inspections). The projects have a limited duration and once finished, they can be integrated to activities, if it is the case.

A schematic chart of the Regulatory Body structure is shown in Figure 8.1.

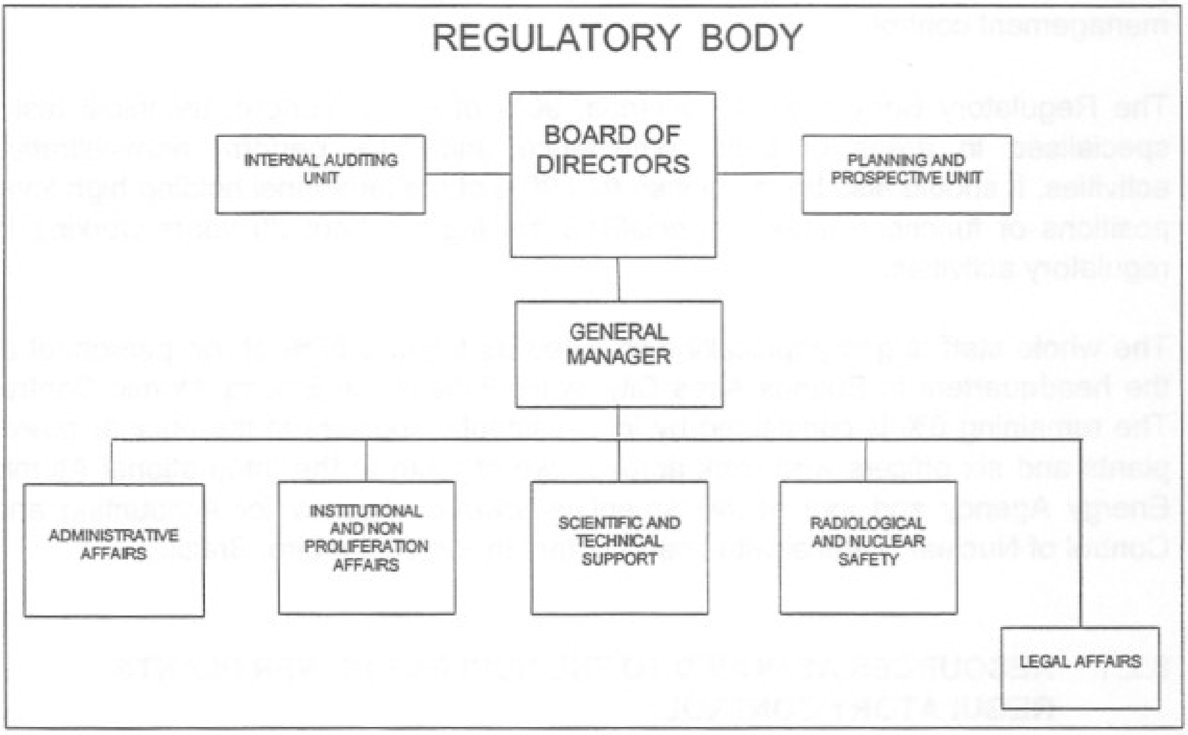


Figure 8.1 – Regulatory Body - Organisation Chart

The General Manager has the main responsibility of leading the executive activities of the Regulatory Body.

The Administrative Affairs Division provides administrative and accounting support to the Regulatory Body regulatory tasks.

The Radiological and Nuclear Safety Division carries out regulatory inspections and assessments concerning radiological and nuclear safety, and technical evaluations of the nuclear power plants licensing process.

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

The Institutional Affairs and Non Proliferation Division controls from the regulatory view point, the use of nuclear materials, equipment and installations of nuclear interest and verifies the compliance with international agreements related to non proliferation guarantees. This branch also controls the compliance of physical protection regulations applicable to nuclear materials and installations. Besides, it co-ordinates the institutional relations in the national and international sphere.

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

The Regulatory Body has 215 persons, 90% of whom perform technical tasks specialised in areas of their competence and 10% perform administrative activities. It should also be mentioned that 90% of the personnel holding high level positions or functions have a specialised training of about 20 years working in regulatory activities.

The whole staff is geographically distributed as follows: 63% of the personnel at the headquarters in Buenos Aires City, while 31% in the Ezeiza Atomic Centre. The remaining 6% is constituted by four resident inspectors in the nuclear power plants and six officers who work abroad, two of them at the International Atomic Energy Agency and four at the Argentine-Brazilian Agency for Accounting and Control of Nuclear Material with headquarters in Río de Janeiro, Brazil.

8.2.1 RESOURCES ASSIGNED TO THE NUCLEAR POWER PLANTS REGULATORY CONTROL

8.2.1.1 During operation

The distribution of Regulatory Body man-days dedication assigned directly to nuclear power plant inspection and safety assessments during 1998 is shown in Table 8.1. These tasks include two resident inspectors in each nuclear power plant, and the safety analysers, not only those who perform supporting to inspection activities but also those who study particular problems regarding the installation safety.

Table 8.2 shows the distribution of Regulatory Body man-days dedication assigned to projects and activities indirectly related to nuclear power plant safety, and sectors providing the corresponding administrative infrastructure, during 1998.

From both tables it may be concluded that the total number of man-days/year with direct or indirect dedication is approximately 9400. The comparison between this value and the total number of man-days/year corresponding to the four resident inspectors (approximately 900) shows that the ratio between them is approximately one to ten.

Table 8.1 - Human resources dedicated to projects and activities directly related to nuclear power plant safety

PROJECTS AND ACTIVITIES	MAN-DAYS / YEAR (1998)
Inspections and evaluations in power reactors	2398
Support in nuclear safety	407
Study of accidental scenarios	407
Core material behaviour	110
Phenomenology of accidents	834
Dispersion model development	363
Software reliability	198
Probabilistic analysis of the consequences of nuclear accidents	55
Validation CNA-I mock-up	187
PSAPACK adaptation	385
Performance indicators	99
Probabilistic criteria of acceptance	33
Off-site power supply reliability	44
Regulatory use of the probabilistic safety assessment	187
Maintenance and evaluation criteria	110
Incident Reporting System	154
TOTAL	5971

Table 8.2 - Human resources dedicated to projects and activities indirectly related to nuclear power plant safety

PROJECTS AND ACTIVITIES	MAN-DAYS / YEAR (1998)
Dosimetry projects	612
Radiation measurements	224
Environmental studies	369
National safety report	682
Dose records	253
Legal affairs	39
General administration	1275
TOTAL	3454

8.2.1.2 **During construction and commissioning**

The human resources used by the Regulatory Body during CNA-I construction and commissioning stages have been different to those assigned to CNE for the same stages. This was due to the different circumstances in which both projects were developed and to the different Regulatory Body experience in such occasions.

In the case of CNA-I the role of Independent Authorised Inspector, prescribed by the ASME code, was performed by two entities: *Technischer Üeberwachungs Verein, Baden* (TÜV), appointed by Siemens Company and *Control e Inspección de Seguridad de Centrales Nucleares* (CISIN -Nuclear Power Plants Safety Control and Inspection) on behalf of National Atomic Energy Commission. Such groups carried out the verifications of preliminary tests, material reception, tests of component, equipment and safety systems fabrication and functioning.(see Section 14.1.1)

In the case of CNE, the Regulatory Body organised a special committee, called Executive Committee for CNE Licensing, in order to co-ordinate the tasks related to CNE licensing. The main functions of this committee, which carried out its activities during the nuclear power plant construction and commissioning, were: to analyse the Safety Report, the commissioning program, the quality assurance program, to perform or require the performance of a safety analysis, to carry out inspections and audits and make requirements to the Responsible Organisation. This committee performed several safety evaluations during the plant construction and commissioning on its own or by contract with third partners. The seismic re-evaluation of the installation is significant among them. (see Section 14.1.2)

8.2.2 **REGULATORY BODY PERSONNEL QUALIFICATION**

As part of their training, the Regulatory Body professional personnel shall attend a Post Graduate Course on Radiological Protection and Nuclear Safety. Subsequently, those professionals assigned to nuclear power plant inspections are trained in research reactors, perform practices in a full scope simulator and finally receive a wide and complete specific training in nuclear power plants. On the other hand, inspectors also participate in periodic courses and seminars organised by these installations for their own personnel as well as in courses and seminars organised by different domestic and international organisations.

The Post Graduate Course on Radiological Protection and Nuclear Safety began in 1977 and since 1981 it was dictated annually, with the co-operation of the University of Buenos Aires, the Ministry of Public Health and the International Atomic Energy Agency.

Besides, the Regulatory Body specialised personnel has participated in activities of organisations such as the International Commission on Radiological Protection, the United Nations Scientific Committee on the Effects of Atomic Radiation and the International Atomic Energy Agency, as well as in OSART missions in different foreign nuclear power plants.

8.3 FINANCIAL RESOURCES

The effective fulfilment of the regulatory objectives requires for the Regulatory Body an efficient structure and adequate personnel together with the necessary economical resources. Concerning this matter Act No 24804, 1997 in its Article 25, establishes that such resources shall be basically obtained from the following incomes:

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

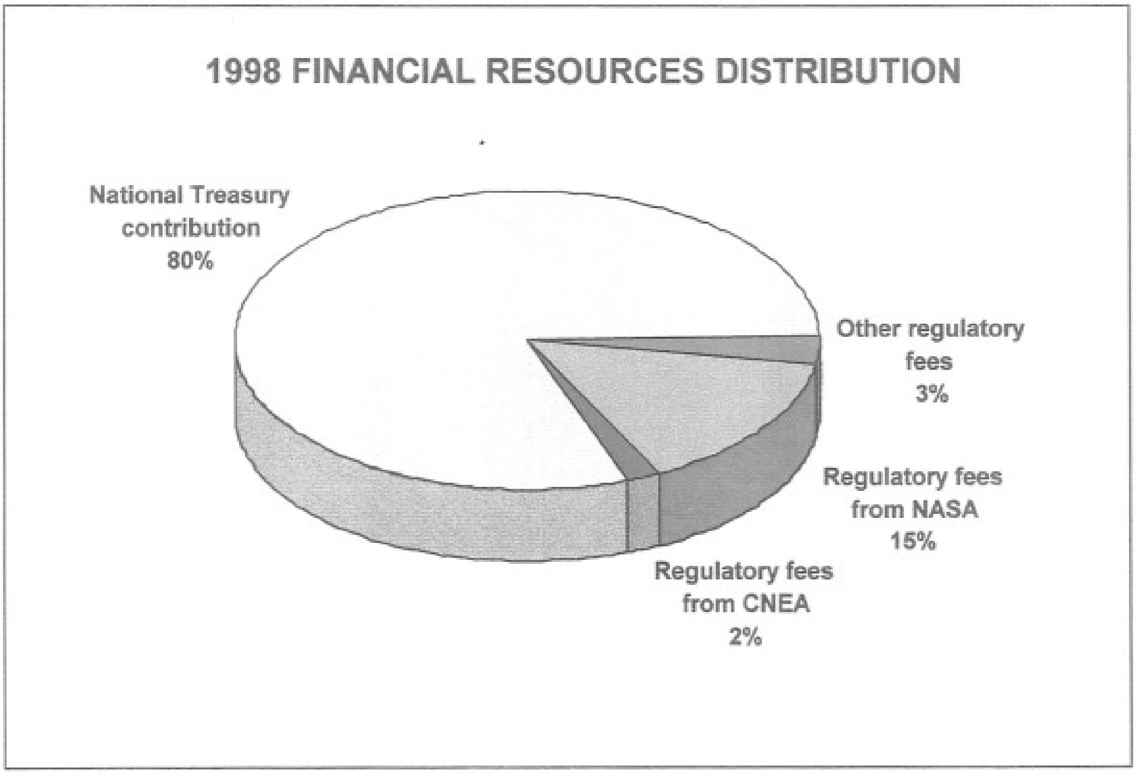
For the case of nuclear power plants under construction or operation, Article 26 of the mentioned act sets the amount of the annual regulatory fees, as a function of the nominal installed nuclear power, that must be annually paid by the Responsible Organisation until the end of the tasks concerning the withdrawal of irradiated fuel elements from the core during decommissioning.

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

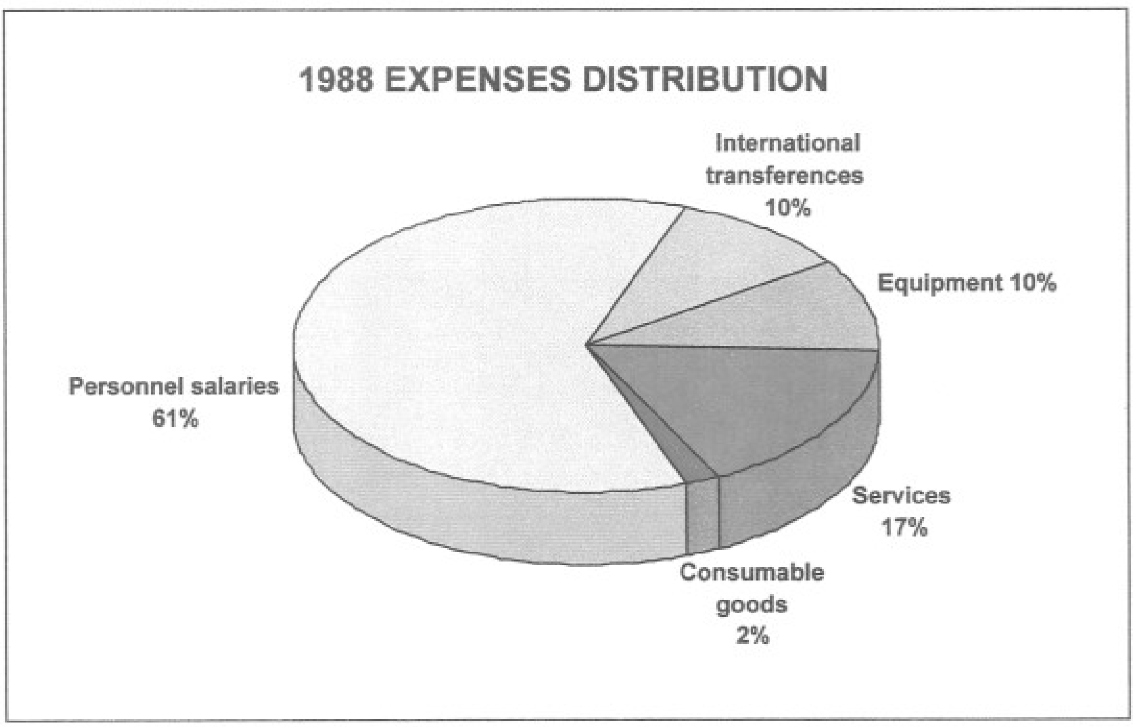
The budget assigned to the Regulatory Body for the financial year 1998 was USD 18,622,065 as detailed in Table 8.3. Additionally, in Graphs 8.1 and 8.2 both the financial resources distribution and expenses distribution for 1988 are shown.

Table 8.3 - Regulatory Body budget for financial year 1998

Expenses and Investment	Item	Treasury contribution (USD)	Own resources (Fees) (USD)	Total (USD)
Expenses	Salaries	10,588,770	767,280	11,356,050
	Consumer goods	84,000	370,000	454,000
	Services	884,065	2,174,950	3,059,015
	Transference to international agencies	1,453,000	189,815	1,642,815
	Fellowships	0	210,185	210,185
Investments	Equipment	1,900,000	0	1,900,000



Graph 8.1 – Financial resources distribution



Graph 8.2 - Expenses distribution

8.4 RELATIONSHIP WITH OTHER ORGANISATIONS

While performing its regulatory functions, the Regulatory Body keeps an active interaction with several national, governmental and private institutions, with the purpose of promoting experience and information exchange, and developing technical co-operation with them. For example the participation of the Nuclear Regulatory Authority in the Forum of Ibero-American Nuclear Regulator and in the Network of Regulators of Countries with Small Nuclear Programs.

Nevertheless, the Regulatory Body is independent of any organisation dedicated to the use or the promotion of nuclear energy, in any of its forms and, as it has already been mentioned, has its own resources for the fulfilment of its mission.

The relationship between the Regulatory Body and a wide variety of domestic and foreign organisations is carried out through agreements which rule the co-operation provided by such institutions of acknowledged technical-scientific level and independent criteria.

A list of the existing agreements detailing their respective purpose, is shown in Table 8.4 for domestic organisations, and in Table 8.5 for foreign organisations.

8.5 COMPLIANCE WITH THE OBLIGATIONS IMPOSED BY THE CONVENTION

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

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

Table 8.4 - Agreements with domestic organisations

AGREEMENT WITH	PURPOSE
Argentine Naval Prefecture *(Prefectura Naval Argentina)*	Co-operation in carrying out verifications of violations of regulatory standards
Clinic Hospital *(Hospital de Clínicas)*	Co-operation in the assistance of persons accidentally overexposed to ionising radiation.
Argentine National Gendarmerie *(Gendarmería Nacional Argentina)*	Studies and advise on subjects related to the prevention of intentional acts that could lead to eventual radiological consequences as well as to the prevention of robbery, theft or unauthorised removal of protected material.
INVAP S.E.- Argentine technology organisation dedicated to high technology projects (*Investigaciones Aplicadas Sociedad del Estado*)	Formal opinion of Nuclear Regulatory Authority concerning the licensability of the installations under construction by INVAP in Egypt.
University of Buenos Aires *(Universidad de Buenos Aires)*	Performance of studies and research of common interest. Performance of an annual course on nuclear and radiological safety
Ministry of Environment and Public Works of the Province of Mendoza, Municipality of San Rafael, National University of Cuyo and National Atomic Energy Commission *(Ministerio de Medio Ambiente y Obras Públicas de la Provincia de Mendoza, la Municipalidad de San Rafael, la Universidad Nacional de Cuyo y la Comisión Nacional de Energía Atómica)*	Execution of programs related with environmental monitoring
Argentine Federal Police - Superintendence of Firemen *(Policía Federal Argentina (Superintendencia Federal de Bomberos)*	Harm protection related to radiological and nuclear safety.
National University of Cuyo *(Universidad Nacional de Cuyo)*	Performance of studies, assessment, technological research and development related to the radiological and nuclear safety.
National University of San Juan *(Universidad Nacional de San Juan)*	Co-operation of scientific-technologic nature. Performance of studies and research related to electric system reliability.
Civil Association Ciencia Hoy *(Asociación Civil Ciencia Hoy)*	Co-operation in communication systems.
National Atomic Energy Commission *(Comisión Nacional de Energía Atómica)*	Optimisation of the use of professional and material resources existing in both institutions.
Ministry of Health and Social Affairs ***(Ministerio de Salud Pública y Acción Social)***	Performance of an annual course on nuclear and radiological safety

Table 8.5 - Agreements with foreign organisations

AGREEMENT WITH	PURPOSE
University of McMaster **(Canada)**	Establishment of the conditions to be fulfilled by both parts concerning the donation of a Tandem Accelerator FN by the University to the Nuclear Regulatory Authority.
Nuclear Regulatory Authority **(Armenia Republic)**	Technical co-operation and information exchange regarding nuclear regulation.
Hauptabteilung fur die Sichercheit der Kernanlagen **(HSK, Swiss)**	Information and experience exchange related to nuclear regulatory aspects
Nuclear Regulatory Commission **(NRC, USA.)**	Co-operation and experience exchange in the application of international codes concerning safety analysis of nuclear power reactors
Council for Nuclear Safety **(South Africa)**	Co-operation and technical information exchange related to nuclear safety
National Radiological Protection Board **(NRPB, UK)**	Scientific and technical information exchange of interest for both parts, in the area of radiological protection.
Atomic Energy Control Board **(AECB Canada)**	Technical co-operation and information exchange regarding nuclear regulation.
Nuclear Regulatory Commission **(NRC,USA)**	Technical co-operation and information exchange regarding regulatory subjects.
National Centre for Nuclear Safety and Radiation Control **(NCNSRC, Egipt)**	Technical co-operation and information exchange regarding nuclear regulation.
Brazilian Argentine Agency for Accounting and Control of Nuclear Materials **(ABACC)**	Co-operation in safeguards technique exchange, uses of laboratories, equipment and services of interest for both parts.
Commisariat à l´Energie Atomique **(CEA, France)**	Scientific and technical co-operation in the field of pacific uses of nuclear energy.
Electric Power Research Institute **(EPRI, USA)**	Areas of reliability, verification, validation and licensing of safety programs for nuclear power plants.
National Nuclear Safety Council **(Spain)**	Technical information exchange and co-operation regarding radiological and nuclear safety.
European Commission **(EC)**	Software used in the assessment of nuclear accident consequences.
Department of Energy **(DOE, USA)**	Research and development in nuclear material control, accountancy, verification and physical protection.
Institut de Protection et Sûreté Nucleaire **(IPSN, France)**	Severe accident analyses.

ARTICLE 9

RESPONSIBILITY OF THE LICENSEE

9.1 BACKGROUND

At the beginning of nuclear activity in the country, the installations had neither the complexity nor the characteristics that could make accidents with significant radiological consequences conceivable. The responsibility for radiological safety of such installations was assigned to one person, generally the installation head, who by himself or with the help of his personnel or contracting third part services, carried out all the safety related tasks. The Regulatory Body required that such person should be duly qualified, providing him with the corresponding individual licence or authorisation, which proved his qualification. When the design, construction and pre-operational tests of an installation demonstrated to be satisfactory for the Regulatory Body, the corresponding operation licence or authorisation was granted.

Though these concepts are still essentially valid for smaller installations, several improvements have been introduced to the regulatory system as time passed. Thus, when the operational characteristics of installations make it advisable, the Regulatory Body requires that those persons holding certain positions in the operation chart staff receive specialised training and have their own individual licence. Besides, the training requisites for the whole operation personnel were increased (see Section 7.3.2.2).

On the other hand, for the case of a nuclear power plant, the Regulatory Body considered that it was not sufficient to have enough and suitably trained personnel to guarantee its operation as safely as it was originally designed; obviously, technological progress demands a periodic review of design and operation aspects in such kind of installations and, if corresponds, the introduction of those modifications the state of the art of safety makes advisable. These considerations led to the creation of the figure of the Responsible Organisation.

9.2 RESPONSIBLE ORGANISATION AND PRIMARY RESPONSIBLE

The Regulatory Body requires that each nuclear power plant is sustained by an organisation capable of providing its personnel with the necessary support for the fulfilment of those tasks related to radiological and nuclear safety, such as the revision of operation procedures, maintenance of safety systems, technical modifications of the plant, etc. The organisation is known as Responsible Organisation (Nucleoeléctrica Argentina S.A. NASA - Company in charge of the operation of the nuclear power plants). The standards AR 0.0.1 and AR 10.1.1 establish its responsibilities, being some of the significant ones:

- The Responsible Organisation should do whatever reasonable and compatible with its possibilities regarding safety, fulfilling at least with standards and requirements issued by the Regulatory Body. Such responsibility extends to the stages of design, construction, commissioning operation and decommissioning of the nuclear power plant.
- The fulfilment of regulatory standards and procedures is a necessary but not sufficient condition concerning the Responsible Organisation responsibility, which must do whatever reasonable and compatible with its possibilities regarding safety. Besides, it shall follow the standards and obligations imposed by other competent bodies not related to radiological aspects (e.g. conditions for conventional discharge of chemical effluents).
- The Responsible Organisation may support the operation of more than one installation and delegate the execution of tasks totally or partially, but it maintains the whole responsibility.
- In each nuclear power plant the Responsible Organisation shall appoint a person of its own body, named Primary Responsible, who will be assigned the direct responsibility for the radiological and nuclear safety of such installation, as well as for the fulfilment of standards, licences and requirements applicable to it. In the case of nuclear power plants in operation, their manager is usually the respective primary responsible.
- The Responsible Organisation must provide the necessary support to the Primary Responsible in order to allow him to perform his task and responsibility, and must supervise him to verify that he carries out his responsibility satisfactorily.
- The Responsible Organisation shall submit to the Regulatory Body the technical documents needed to evaluate the safety of the nuclear power plant which the operation licence is applying for.
- No modification of a nuclear power plant changing either design, operational features or mandatory documentation contained in the operating licence, can be initiated without previous Regulatory Body authorisation.
- Both, the Responsible Organisation and the Primary Responsible must facilitate the performance of regulatory inspections and audits, every time the Regulatory Body requires it.
- Every change in the Responsible Organisation structure, that could affect its capability of carrying out its responsibilities, shall be previously approved by the Regulatory Body.

Moreover, the Responsible Organisation must assume the civil responsibility than the Vienna Convention on Civil Responsibility for Nuclear Damages (ratified by Act No 17048,1966) has determined for the licensee. Act No 24804,1997 establishes that the Responsible Organisation is responsible up to a sum of 80 million US dollars for a nuclear accident, being the State responsible for the remaining responsibility.

Apart from the responsibilities of both the operating organisation and the primary responsible of a nuclear power plant, the Regulatory Body has delimited the responsibilities of workers. In relation with this aspect, the AR 10.1.1 standard establishes that workers are responsible for the fulfilment of those procedures elaborated with the purpose of ensuring their own protection, as well as that of other workers and the public. This subject is consistent with the International Atomic Energy Agency recommendations.

9.3 REGULATORY CONTROL ON THE FULFILMENT OF THE LICENSEE'S RESPONSIBILITIES

Since 1958, the Regulatory Body controls the fulfilment of standards, licences and authorisations granted. In order to verify if the licensees fulfil the corresponding responsibilities, the Regulatory Body carries out different types of controls, detailed as follows:

- The Regulatory Body has a constantly updated information of the installations operation chart. The operation licence sets that any modification to the organisation chart must be reported to the Regulatory Body thirty days before the date of execution, being this requirement fulfilled. Besides, these modifications are usually known by the Regulatory Body either through the routine meetings held with the Responsible Organisation or via the resident inspectors report.
- The AR 0.11.1 standard sets the requisites to be fulfilled by the nuclear power plant personnel in order to obtain the corresponding individual licences and specific authorisations, according to Section 7.3.2.2.
- The procedure of issuing individual licences and specific authorisations allows the Regulatory Body to control the aptitude of those persons who must assume responsibilities concerning safety. This aptitude is again evaluated when the specific authorisation is renewed.
- The individual licence may be suspended or cancelled by the Regulatory Body if a lack of any condition demanded for such licence is observed during the performance of tasks. In the same way, the specific authorisation may also be modified, suspended or cancelled according to what is expressed in Section 7.4, particularly in its last paragraph.
- Besides, the Regulatory Body carries out a permanent verification that the Primary Responsible fulfils the responsibilities related to safety, and particularly the requirements emerging from the applicable standards, the operation licence conditions and any other requirement related to radiological safety. This is done

through regulatory audits and inspections carried out by resident inspectors and analysts of the Regulatory Body.

- The Regulatory Body also verifies that the Responsible Organisation fulfils its main responsibilities related to safety.
- Moreover, the Regulatory Body performs a permanent follow up of the Technical Revision Committee and the Safety Advisory Internal Committee minutes (see Section 10.2.1).
- The Regulatory Body has also issued a sanctions regime for the case of non fulfilment of any of the regulatory requisites, according to what is expressed in Section 7.4, particularly in its last paragraph.

9.4 COMPLIANCE WITH THE OBLIGATIONS IMPOSED BY THE CONVENTION

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

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

ARTICLE 10

PRIORITY TO SAFETY

10.1 GENERAL PRINCIPLES

Since the beginning of nuclear activities in the country, the State has considered that nuclear power plant safety should be of top priority (also extended to other installations or practices) during their stages of design, construction, commissioning, operation and decommissioning.

This priority to radiological and nuclear safety is made clear in that such stages shall be carried out in accordance with a coherent regulatory system of principles, criteria and safety policies applied in the country since some decades ago, being even occasional contributions that the country made to international organisations such as IAEA and ICRP.

Two of the principles the regulatory system must comply with are: the principle of regulatory control and that of responsibility for safety.

Two organisations are involved in the compliance with the above mentioned principles: the Regulatory Body in the case of the regulatory control principle, and the Responsible Organisation in the case of the responsibility for safety principle. Thanks to both principles, these two institutions coexist being, at the same time, completely independent from each other.

The Nuclear Regulatory Authority establishes and applies a regulatory frame to all the nuclear activities developed in the country, with the following purposes:

- Protect people against harmful effects of ionising radiation.
- Keep watch over radiological and nuclear safety in the nuclear activities developed in the country.
- Make sure that nuclear activities are not developed with non authorised purposes according to Act No 24804, 1997 "National Law of Nuclear Activity", the rules that may be consequently dictated, international agreements and non proliferation policies adopted by the country.
- Prevent intentional acts that may cause severe radiological consequences or the unauthorised removal of nuclear materials or other materials and equipment of nuclear interest subject to regulation and control according to Act No 24804, 1997.

These purposes are compatible with the global strategy of the regulatory system, which aims particularly at the following basic aspects (as expressed in Section 8.1):

- Regulatory inspections and audits for the verification of the compliance with licenses and authorisations issued.
- Independent performance of studies and assessments about radiological and nuclear safety, safeguards and physical protection.
- Scientific and technological development in subjects related to radiological and nuclear safety, safeguards and physical protection.
- Personnel training in subjects related to radiological and nuclear safety, safeguards and physical protection, for those members of the staff who are responsible for the safety of radiological practices subject to control, and for those who perform regulatory activities.

As regards the Responsible Organisation and from the point of view of safety (as shown in the report *Políticas y Principios de Nucleoeléctrica Argentina S.A.* - Policies and Principles of the Company in charge of the operation of the nuclear power plants) its course of action is such that:

- It complies with pertinent regulatory standards and requirements and performs, in addition, all what is reasonable and compatible with its possibilities in favour of radiological and nuclear safety in nuclear power plants, concerning their design, construction, commissioning, operation and decommissioning. To that respect, and according to nuclear power plant operation, the before mentioned organisation takes the following documents into account:
    * Safety Report.
    * Operating Manual.
    * Policies and Principles Manual.
    * Maintenance Manual.
    * Quality Assurance Manual.
    * Radioprotection Procedures (Code of Practice).
    * In-Service Inspection Program.
    * Periodic Test Program.
    * Emergency Plan.
    * Personnel Qualification and Training Program.
    * Operating License.
- It improves the existing safety practices continuously.
- Ensures that those guides accepted and adopted by the nuclear industry are being fulfilled, when applicable for the case of domestic nuclear power plants.
- It sustains an attitude towards safety based on the organisation self-evaluation, the feedback of operative experience, technological development and the early prevision of possible degradation of the plants that might affect their safety.
- It continuously carries out training and retraining courses for the plant personnel or for those members of the staff who perform safety related tasks.

Finally, it should be mentioned that the regulatory system also complies with the concept of safety culture, which implicitly results from the before mentioned criteria regarding the Regulatory Body and the Responsible Organisation.

From the preceding considerations, it should be clearly noticed that the regulatory system (from the point of view of both the Regulatory Body and the Responsible Organisation) guarantees the prioritisation of radiological and nuclear safety in nuclear power plants, in what concerns their design, construction, commissioning, operation and decommissioning.

10.2 SPECIFIC ACTIONS

10.2.1 SAFETY POLICY

The safety principles before described are fulfilled in every activity related to nuclear power plants. In particular, the priority to safety may be noticed in the operating license and in the policies and principles manual for each installation, including the fact that there are limits and operating conditions for any of the nuclear power plants considered.

The Regulatory Body establishes in the operating license that two committees must exist to advise the nuclear power plants managers (Primary Responsible) and the highest Responsible Organisation authorities concerning safety subjects; these committees are:

- Safety Advisory Internal Committee.
- Technical Revision Committee.

The Safety Advisory Internal Committee reports to the plant manager and its members, selected because of their knowledge and experience, being 60% installation personnel, and the rest belonging to other sectors of the Responsible Organisation. The committee advises the Plant Manager regarding the actions to be followed in, for instance:

- Outages.
- Safety related incidents.
- Modifications to the installation (safety or safety related systems).
- Abnormal situations.
- Periodic evaluation of the installation performance.
- Periodic evaluation of the training personnel program.
- Emergency plan, etc.

The advice given by Safety Advisory Internal Committee consists of analyses, conclusions and recommendations issued as a minute signed by its members.

On the other hand, the Technical Revision Committee, which is independent of the installation, advises the maximum staff level of the Responsible Organisation as regards the safe operation of the nuclear power plants, analyses the importance of the failures, abnormal and significant events, evaluates the proposed design modifications that may affect the safety related systems, and in general about the same subjects dealt with by the Safety Advisory Internal Committee.

The Technical Revision Committee is integrated by acknowledged outstanding professionals of the Responsible Organisation, appointed because of their knowledge and experience constituting the Responsible Organisation technical support, and its own conclusions and recommendations are issued as minutes.

Both committees' minutes are of the highest importance for the Regulatory Body, due to the fact that both the Primary Responsible and the Responsible Organisation independently produce a written evidence of their opinion concerning nuclear safety related subjects referred to the particular installation considered.

10.2.2 SAFETY CULTURE AND ITS DEVELOPMENT

It has already been said that both the Regulatory Body and the Responsible Organisation have adhered to the safety culture principle, which may be observed in the criteria sustaining the regulatory system.

Thus, and for the case of nuclear power plants, the way the Responsible Organisation acts, not only in what concerns compliance with pertinent regulatory standards and requirements (i.e. compliance with mandatory documents) but also in the effort in carrying out all what is reasonable and compatible with its possibilities in favour of safety, shows the accomplishment of the mentioned principle.

Moreover, the contents of the regulatory documents issued by the Regulatory Body, such as standards, requirements and operating licenses, show, among other aspects, those concepts characteristic of the safety culture principle.

In the same way, the active participation of members of both the Regulatory Body and the Responsible Organisation in several committees and experts meetings organised by IAEA on subjects related to the development of the safety culture, also show the importance the concept is given to in those organisations.

10.2.3 **COMMITMENT TO SAFETY**

The commitment to nuclear power plants' safety is made clear in design or operation concepts giving priority to safety over economic rentability of the mentioned installations.

As clear examples of such commitment it should be mentioned that the nuclear power plants comply with the defence in depth principle (see Section 18.2), or in the case of operational situations in which an apartment from normal operating conditions occurs, and the decision is taken to shutdown the plant (rule that has been observed along the plants' lifetime).

The compliance with Maintenance Programs, In-service Inspection Programs and good operation practices are also part of the commitment.

The commitment to safety is also verified through the assignment of resources for programmed revisions, and for maintaining and improving systems and components related to the plant safety. An actual case is the already mentioned updating and backfitting program in CNA-I (see Section 6.2.1).

10.2.4 **MANAGEMENT ATTITUDES TOWARDS SAFETY**

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

It may be mentioned as well that a follow up is carried out by the highest level of authority both of the installation and the Responsible Organisation, concerning the conclusions and recommendations emerging from Safety Advisory Internal Committee and Technical Revision Committee meetings. In these meetings, the performance on-the-job of nuclear power plant personnel is evaluated, among other activities, and their improvement is encouraged with the help of the conclusions emerging from the analysis done.

It should be mentioned that periodic meetings at the highest level are carried out between the Regulatory Body and the Responsible Organisation. In such meetings the main safety aspects arisen from regulatory inspections, safety analysis and other assessments are considered. Safety aspects related to nuclear power plants programmed outages, as well as the progress in backfitting related activities in the installations are also analysed.

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

10.2.5 PERSONNEL MOTIVATION

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

The attitudes and policies followed by the managing staff concerning personnel thus tend to reinforce the means of encouraging such adherence, that is: initial training, periodic retraining, exchange of (operational, maintenance, etc) experience, professional prestige (both in their own institution and in the national or international nuclear community) and the preservation or even improvement of working position (concerning both the technical and pecuniary aspects).

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

10.2.6 REGULATORY CONTROL ACTIVITIES

One of the regulatory and control functions of the Regulatory Body aims to protect persons against the hazardous effects of ionising radiation, and to preserve the radiological and nuclear safety in the nuclear activities carried out in the country.

This is only possible if there is an actual priority to safety in the regulatory control, priority that may be verified by noticing that the country regulatory management is carried out according to the following aspects:

- Independence of technical opinions and decisions.
- Administrative autarchy.
- Legal capacity to act in the field of public and private rights.
- Qualified personnel.

On the other hand, the Regulatory Body is independent from other organisations related to the operation, distribution, or promotion of power generation. It should be noticed that the Regulatory Body annually reports its activities to the Executive Power as well as to the National Congress.

Finally, it is noted that Article 8 of Act No 24804, 1997 ratifies the precedent considerations, particularly those referred to the safety priority, and the Article 16 of such Act establishes that it is mandatory to send to the Executive Power and to the National Congress a report containing the annual activities and suggestions of measures to be adopted for public benefit.

10.2.7 VOLUNTARY ACTIVITIES AND GOOD PRACTICES RELATED TO SAFETY

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

- Consults and meetings of the Responsible Organisation and Regulatory Body specialists aiming at facilitating and improving the compliance with general and specific requirements, evaluating, in addition, the operational situation of the plant.
- Participation in the International Atomic Energy Agency Incident Reporting System, that enables the contribution and return of operational experience, from which some actions may be applied to the domestic nuclear power plants.
- Active participation of the Responsible Organisation in international organisations of nuclear operators: the Candu Owners Group and the World Association of Nuclear Operators. Both organisations promote the exchange of operational experience and give technical assistance as response to nuclear power plants requests.
- Implementation of external technical audits, for instance the peer review performed by the World Association of Nuclear Operators to both nuclear power plants, IPERS mission to CNA-I and the OSART mission performed by IAEA to Embalse Nuclear Power Plant.
- Interaction with official and non-governmental bodies, with the purpose of analysing emergency preparedness measures, including the role of the Regulatory Body and other organisations (i.e. Civil Defence at national level).

10.3 COMPLIANCE WITH THE OBLIGATIONS IMPOSED BY THE CONVENTION

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

ARTICLE 11

HUMAN AND FINANCIAL RESOURCES OF THE LICENSEE

11.1 INTRODUCTION

As already mentioned in Section 6.1, CNA-I and CNE nuclear power plants were operated by the National Atomic Energy Commission until August 1994, when the Executive Power, according to its decision of transferring nuclear power generation activities to the private sector, created the company named Nucleoeléctrica Argentina S.A. (NASA - Company in charge of the operation of the nuclear power plants-) through the Decree No 1540, 1994.

According to the previsions in Article 4 of the mentioned decree, this company 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, according to the Regulatory Body applicable standards in the related subjects, and those regulating the wholesale electrical market.

The staff belonging to the former Nuclear Power Management Division, and to ENACE (Argentine Nuclear Company of Electrical Power Plants), together with personnel coming from other National Atomic Energy Commission sectors, was transferred to the new company.

The objectives of the company are:

- To generate nuclear electric power.
- To improve safety levels continuously.
- To increase load factors and plant availability with competitive costs as compared with the other generators in the wholesale electric market.

In the Act No 24804, 1997 it was established that the company must comply with in force standards regarding radiological protection and nuclear safety issued by the Regulatory Body, safeguards agreements subscribed by the country as well as any obligation emerging from other conventions.

Both, the decree of constitution and Act No 24804, 1997, establish that Nucleoeléctrica Argentina S.A. shall be privatised according to the transformation of public services companies. In the meantime, the total company stock is owned by the State, being the Energy Secretary of the Ministry of Economy and Public Works and Services responsible for the exercise of the society rights.

11.2 LICENSEE ORGANISATIONAL STRUCTURE AND FINANCIAL AND HUMAN RESOURCES

11.2.1 LICENSEE ORGANISATIONAL STRUCTURE

The organisation of Nucleoeléctrica Argentina S.A. is constituted by a Directorate having three directors who, in turn, act as President, Vice-President and General Manager. As usual in this type of society, the internal Auditing Unit and the Legal Affairs Department report to the Directorate as shown in Figure 11.1

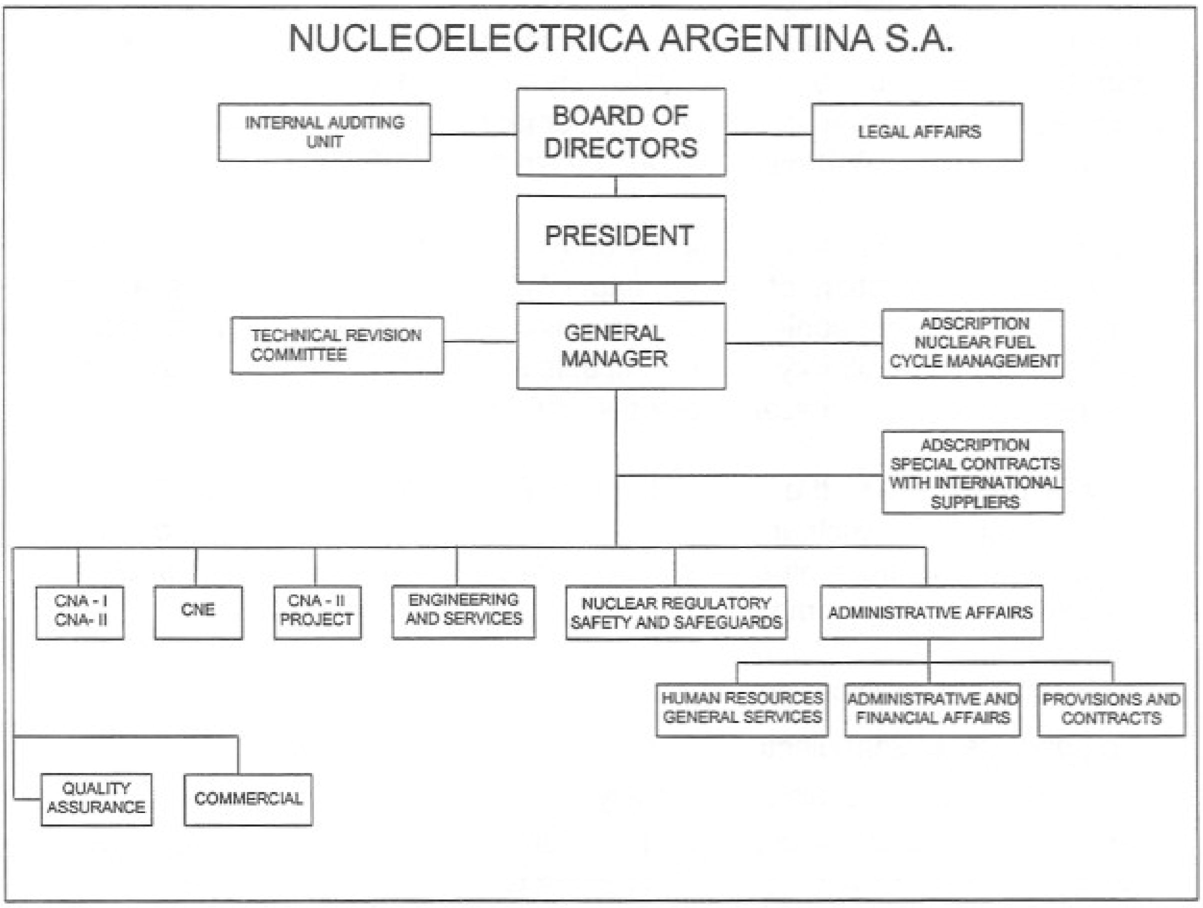


Figure 11.1 - Licensee - Organisation Chart

Six divisions report to the General Manager. Among these divisions, CNA-I, CNA-II and CNE should be mentioned. CNA-II Division comes from the Argentine Nuclear Company of Electrical Plants, presently dissolved. The Technical Revision Committee and the Quality Assurance Department also report to the General Manager.

The Engineering and Services Management Section, also reporting to the General Manager, provides technical support.

The Nuclear Regulations, Safety and Safeguards Section acts as an interface between Nucleoeléctrica Argentina S.A. and the Regulatory Authority, and as technical support to the Responsible Organisation in aspects concerning Radiological Protection, Nuclear Safety, Safeguards and Physical Protection.

Nucleoeléctrica Argentina S.A.'s organisation is completed with sections in charge of commercialising the generated electric energy, preparation of special contracts, fuel management, human resources, budget and finances, and provision of supplies and services.

11.2.2 **LICENSEE HUMAN RESOURCES**

The whole Nucleoeléctrica Argentina S.A. staff is 1076 people, of whom 279 are professional, 654 are technicians and 143 are administrative. There are 921 people working at the nuclear power plants while 155 work at the headquarters. Distribution is shown in Table 11.1

Table 11.1- Responsible Organisation staff

	CNA-I	CNA-II	CNE	RESPONSIBLE ORGANISATION	TOTAL
PROFESSIONALS	82	34	83	80	279
TECHNICIANS	230	5	386	33	654
ADMINISTRATIVES	31	3	67	42	143
TOTAL	343	42	536	155	1076

The number of people working at nuclear installations is variable due to the fact that during programmed outages and short shutdowns, temporary personnel is required with technical capability to work in maintenance tasks. The personnel of supplying companies involved in highly specialised tasks required at the installations, is not included here. Figures 11.2 and 11.3 show the organisation charts of both CNA-I and CNE nuclear power plants.

11.2.3 **LICENSEE CURRENT AND CAPITAL EXPENSES**

The annual budget of Nucleoeléctrica Argentina S.A., as well as the income, is estimated taking into consideration the net electric generation corresponding to the same period. For 1998 the estimated generation values of CNA-I and CNE nuclear power plants are 2.2 and 4.4 TWh respectively. The values corresponding to previous years for both nuclear power plants together are shown in Table 11.2

Table 11.2- Operation Performance for both nuclear power plants

OPERATION PERFORMANCE	1995	1996	1997
Net generation (TWh)	6.56	6.92	7.44
Load factor (%)	80.3	84.5	90.4
Installed nuclear power percentage with respect to total electric generating capacity (%)	5.9	5.9	5.5
Annual generation percentage with respect to total electric generation (%)	11.8	11.7	11.5

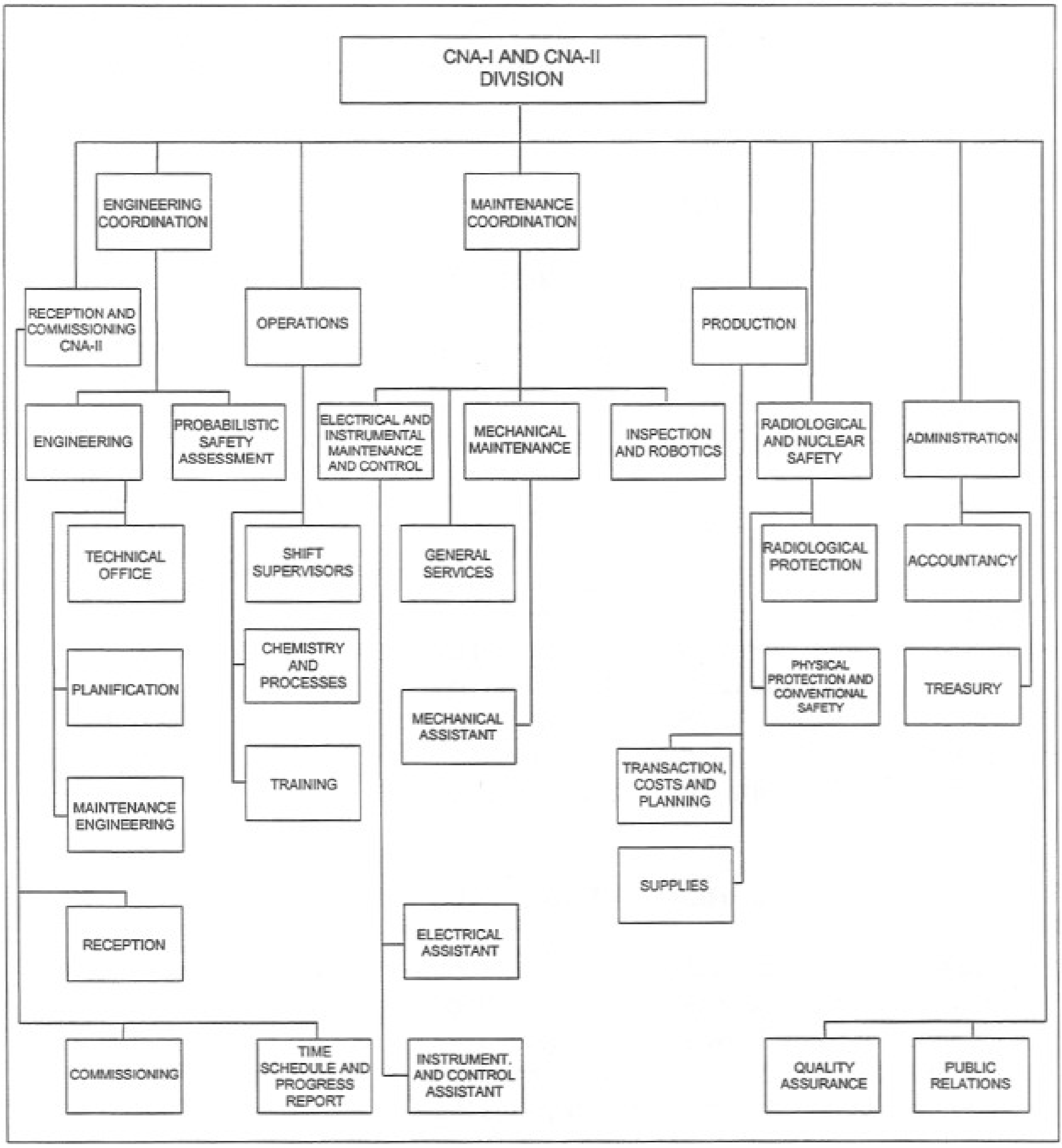


Figure 11.2 - CNA-I and CNA-II Organisation Chart

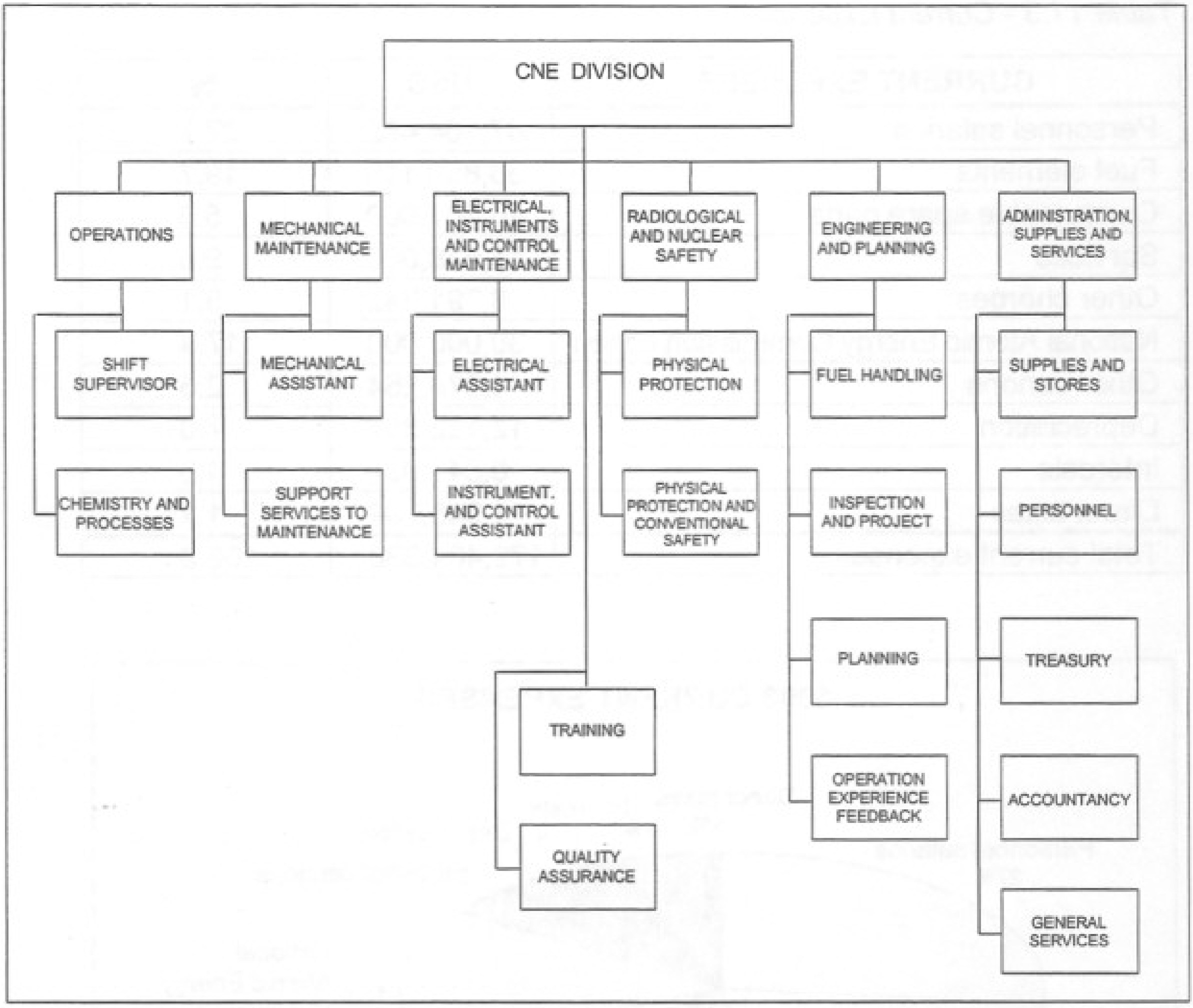


Figure 11.3 - CNE Organisation Chart

For 1998 the company budget, shown in Table 11.3 and Graph 11.1, are discriminated as follows:

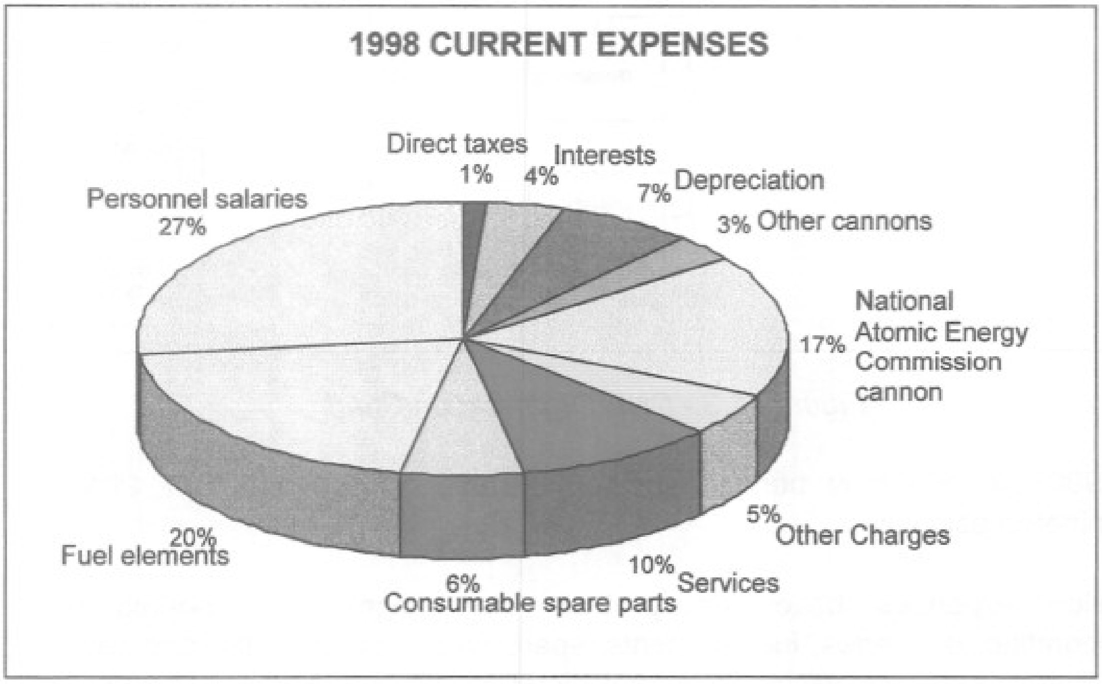
Operation expenses have among their most significant amounts those corresponding to salaries, fuel elements, spare parts, consumables and services. Other Expenses comprise interests and direct taxes. The addition of both items constitutes the so called Current Expenses.

Besides, Capital Expenses, that is to say, that portion of the Current Expenses considered as investment, are shown in Table 11.4 and Graph 11.2. This item also includes backfitting expenses for both nuclear power plants and construction expenses of CNA-II. The most significant item corresponds to CNE's heavy water. It corresponds to restore the heavy water to Atomic Energy of Canada Limited, initially hired to CNE through a contract. Once it is completely restored to Atomic Energy of Canada Limited, Nucleoeléctrica Argentina SA will own the heavy water present at the installation.

It should be made clear that the budget for 1998 only includes the credits obtained in order to finance the purchase of heavy water to the providing company.

Table 11.3 - Current Expenses

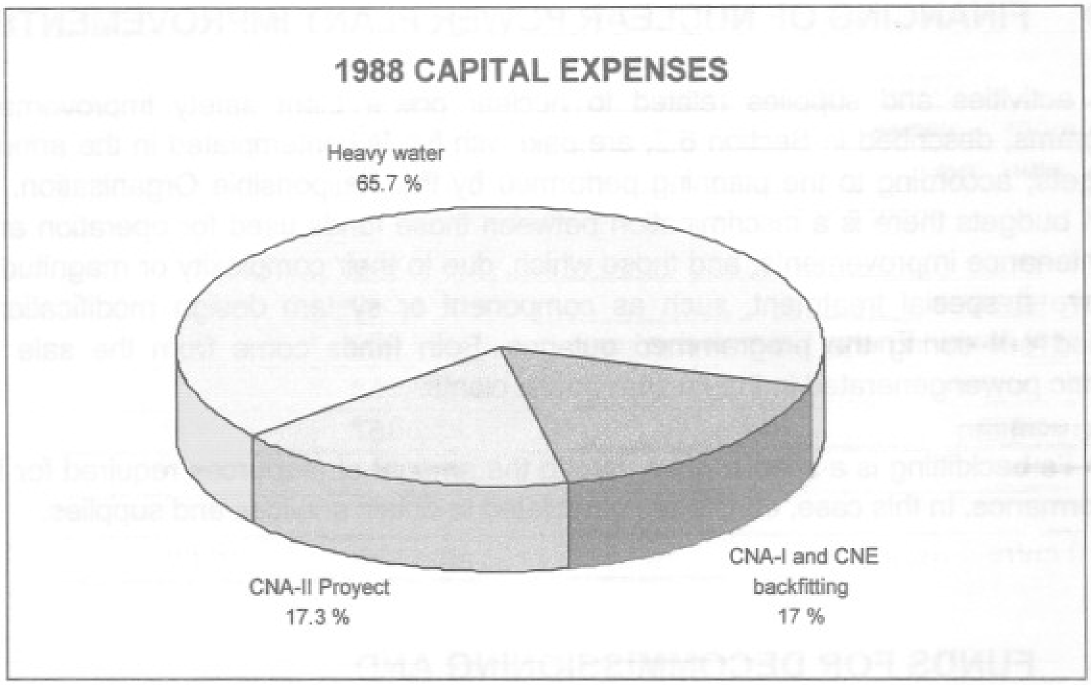
CURRENT EXPENSES	USD	%
Personnel salaries	47,764,492	27.7
Fuel elements	33,850,110	19.7
Consumable spare parts	10,167,000	5.9
Services	16,528,000	9.6
Other charges	8,791,242	5.1
National Atomic Energy Commission canon	30,000,000	17.4
Other canons	4,376,254	2.5
Depreciation	12,132,357	7.0
Interests	6,849,084	4.0
Direct taxes	1,945,799	1.1
Total current expenses	172,404,338	100.0



Graph 11.1 - Current Expenses 1998

Table 11.4 - Capital Expenses for 1998

CAPITAL EXPENSES 1998	USD	%
CNA-I and CNE backfitting	14,214.000	17.0
CNA-II project	14,458,000	17.3
Heavy water purchase	55,000,000	65.7
Total capital expenses	83,672,000	100.0



Graph 11.2 - 1988 Capital Expenses

11.2.4 LICENSEE ECONOMIC RESOURCES

The resources needed by Nucleoeléctrica Argentina S.A. in order to afford its Current Expenses are provided by two sources: the income coming from energy and power sale to the wholesale market and an annual variable contribution from the Unified Fund, the assignment of which is decided by the Energy Secretary. The distribution is shown in Table 11.5

Table 11.5 - *Current Incomes*

Current Incomes	USD	%
Operation Incomes	144,404,899	81.9
Contribution from Unified Fund[1]	31,950,000	18.1
TOTAL CURRENT INCOMES	176,354,899	100.0

As the energy is sold to a market choosing the seller according to the marginal generation cost, the fare received by Nucleoeléctrica Argentina SA may vary as a function of different factors. For 1998 a monomic value of 21.87 mils/kWh is estimated. The contribution of the unified fund for 1998 is 31,950,000 USD, that is to say, about 18% of the total income.

[1] Resources coming from the Energy Secretary of the Ministry of Economy and Public Work and Services to finance exceptional expenses.

11.3 FINANCING OF NUCLEAR POWER PLANT IMPROVEMENTS

The activities and supplies related to nuclear power plant safety improvement programs, described in Section 6.2, are paid with funds contemplated in the annual budgets, according to the planning performed by the Responsible Organisation. In such budgets there is a discrimination between those funds used for operation and maintenance improvements, and those which, due to their complexity or magnitude, require a special treatment, such as component or system design modifications carried out during the programmed outages. Both funds come from the sale of electric power generated in the nuclear power plants.

CNA-I's backfitting is a special case, due to the amount of resources required for its performance. In this case, credits are negotiated to obtain services and supplies.

11.4 FUNDS FOR DECOMMISSIONING AND RADIOACTIVE WASTE DISPOSAL

According to Act No 24804, 1997 in its Article 2, the State assumed the responsibility of radioactive waste management through the National Atomic Energy Commission, and the waste generators must provide the required resources for such management. Moreover, the waste generator is the responsible for its safe storage until its transfer to the National Atomic Energy Commission.

At present, an act proposal referred to radioactive wastes named "Regime of Radioactive Waste Management" is under consideration of the National Congress.

The proposal also establishes the creation of a Management and Final Waste Disposal Fund, to be used in financing the National Program of Radioactive Waste Management in charge of the National Atomic Energy Commission

On the other hand, Act No 24804, 1997 in its Article 9, sets that any nuclear power plant operator must contribute to the nuclear power plant decommissioning fund.

11.5 LEGAL DOCUMENTS RELATED TO TRAINING AND LICENSING OF NUCLEAR POWER PLANT PERSONNEL

The Regulatory Body requires the personnel of nuclear power plant to be properly trained and qualified according to their functions. Besides the personnel performing safety related functions shall be licensed. The applicable Regulatory Standards are AR 0.11.1 and AR 0.11.2.

Both standards set the criteria and procedures to obtain individual licenses and specific authorisations. Moreover, they establish the terms and conditions according to which the Regulatory Body, after requiring the corresponding analysis and reports from its Advisory Councils, can issue such individual licenses and specific authorisations or the renewal of such authorisations. This subject has been widely treated in Section 7.3.2.2.

On the other hand, the document "Regulation for Licenses and Authorisations of Nuclear Power Plant Personnel" sets the program contents for the examinations required to issue the individual licenses and specific authorisations that enable the personnel to hold the corresponding licensable position in the nuclear power plant.

Likewise, the operating licenses of both nuclear power plants set the specific requirements for the re-training of personnel holding licensable positions in the organisation chart. Besides, there are several other regulatory requirements concerning such re-training.

11.6 LICENSEE RESPONSIBILITY IN RESOURCE ASSIGNMENT

It is responsibility of the Responsible Organisation to provide the resources for the installations maintenance and operation. During the operation such resources mainly come from the electric power sale and once the decommissioning is carried out, from the nuclear power plant Decommissioning Fund.

It is of the Responsible Organisation concern to foresee and provide the financial resources to maintain the necessary qualification and training personnel taking into account the corresponding procedures and standards.

When the Regulatory Body approves an installation organisation chart, it sets the minimum necessary staff to operate it as well as its licensable positions. Each applicant shall periodically comply with qualification and training requirements including the psychophysical aptitude test.

11.7 NUCLEAR POWER PLANTS PERSONNEL QUALIFICATION

Both nuclear power plants have a training and qualification section for the whole plant personnel, the responsible of which reports directly to the nuclear power plant manager. Apart from the specific functions related to the basic technical preparation of personnel for each function, such section contributes to safety improvement with programs of continuous execution with the objective of updating knowledge and improving the operational practices.

The following are examples of continuously executing programs: periodic re-training in simulators, safety culture related courses, analysis of operational incidents in seminars, personnel exchange with other nuclear power plants and courses and lectures given by local and foreign specialists.

The qualification sections maintain a fluid relationship with the International Atomic Energy Agency and other operator organisations (World Association of Nuclear Operators and Candu Owners Group), from which they receive as well as transmit information about operational experience.

During last years the qualification programs have been enriched with improvements in both qualification tools and installations. Simulators have been implemented in both nuclear power plants for a more frequent and wide training of operating staff.

The re-training programs are partially based on the operational experience. They comprise external and internal events analyses, including those minor incidents having non-significant consequences.

11.7.1 **ATUCHA I NUCLEAR POWER PLANT**

Important resources have been assigned to the improvement of training facilities. For instance the training building has been renewed including all its classrooms, and a test facility has been built for maintenance personnel training. Such facility is provided with equipment, components, electrical boards and plant devices to ensure an optimal staff training in actual conditions.

Additionally, in fulfilment of a Regulatory Body requirement, a modern graphic interactive simulator has been installed, at present in the validation process.

It should be emphasised that one of the main applications of the probabilistic safety analysis was the incorporation into the operation personnel training program, of specific training in accidental sequences of most influence on the annual frequency value of core damage.

Moreover, the operation personnel have been trained on subjects related to their behaviour in foreseeable severe accident scenarios with high and low primary circuit pressure. Besides, they perform periodic practices in the full scope simulator of the Brazilian Nuclear Power Plant ANGRA-II

11.7.2 **EMBALSE NUCLEAR POWER PLANT**

In the case of CNE, there has been an improvement in the facilities for personnel training, as well as in the number of mock-ups to reproduce the maintenance tasks, specially a complete full scale facility, used to train the staff in the use of the repositioning machine of garter springs and in pressure tube replacement.

The response in the presence of abnormal events is practised in the simulators, and the study of potential accidental sequences will be analysed in the probabilistic safety analysis.

As part of the operation personnel periodic re-training, since several years ago, practices in the full scope simulator of the Canadian Nuclear Power Plant GENTILLY-II are carried out. These practices are a valuable tool for operator training on the diagnosis of normal and accidental situations, and on their behaviour according to abnormal event procedures.

11.8 COMPLIANCE WITH THE OBLIGATIONS IMPOSED BY THE CONVENTION

The analysis of the present Article enables to conclude that in the country the appropriate steps were taken to assign resources in order to preserve safety of each nuclear installation throughout its lifetime.

Therefore, the country complies with the obligations established in Article 11 of the Convention on Nuclear Safety.

ARTICLE 12

HUMAN FACTORS

12.1 SYSTEM TO DETECT, CORRECT AND PREVENT HUMAN ERRORS

The detection, correction and prevention of human errors are carried out through two clearly distinguished processes: the incident analysis and the global and systematic study of the installation safety.

In the first case, the process acts on the abnormal or unexpected events corresponding to actual situations that happen in the installations (operational experience). Such events are unique opportunities to detect and correct human errors, identifying the imperfections regarding organisation, persons, materials and practices. In this case the key elements are the quality of the report on the event occurred, the rigour in the investigation about its deepest causes and the corrective actions carried out.

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

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

It should be pointed out that in both nuclear power plants, the periodic training of the operation personnel in full scope simulators and other simulators existing at the installations, constitutes an additional means to detect, correct and prevent human errors in accidental situations, contributing, in addition, to the improvement of operation procedures (see Sections 11.7.1 and 11.7.2).

On the other hand, standards AR 3.2.1 and AR 3.4.1 establish the information the operator should count with in order to take safety related decisions, the prohibition of interventions during the period immediately after the occurrence of accident initiating events and the characteristics of the man-machine interaction related to the design of the reactor instrumentation and protection systems (see Sections 18.2, 18.3.1 and 18.3.2).

12.2 MANAGEMENT AND INSTITUTIONAL ASPECTS

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

Once such events have happened, the installation Primary Responsible, supported by the Responsible Organisation, determines the responsibility degree, if any, of persons who may have incurred in errors and applies the corrective measures and, if corresponds, the applicable sanctions.

On the other hand, having the Regulatory Body analysed the event, issues requirements and, if it is the case, applies the corresponding sanctions to the Responsible Organisation, the Primary Responsible and the involved personnel.

During the safety inspection and evaluation process of the nuclear power plant, the Regulatory Body pays special attention to find early signals and trends such as:

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

12.3 HUMAN RELIABILITY ANALYSIS

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

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

The data used in human reliability models depend explicitly on the applied model and come from operational experience, generic data and practices in foreign simulators of compatible plants, since in the country there are no nuclear power plant full scope simulators. Specifically, the human reliability analysis carried out for CNA-I probabilistic safety assessment was based on generic data for the human error failure rate, from factors, recovery and uncertainty factors. CNA-I operational experience provided task execution times, actuation frequency for components and equipment, and equipment recovery times.

For the case of CNE probabilistic safety assessment, the feasibility of incorporating information coming from periodic practices CNE operating staff is carrying out at Gentily's full scope simulator, is being evaluated.

12.4 REVISION OF THE CONTROL ROOMS DESIGN

According to the experience acquired at international level, the importance and implications of human errors are significant for the safe operation of nuclear power plants. Taking into account that in the sixties the design of CNA-I control room did not consider human participation in the operation adequately, modifications to the plant were implemented in order to improve such inter-relation. Although CNE has been designed according to a much more modern concept, some modifications have also been carried out in the control room.

12.4.1 IMPROVEMENTS TO ATUCHA I CONTROL ROOM

Improvements in the ergonomic design of CNA-I control room were carried out, both by the inclusion of new devices, or in the display system of safety parameters and the significant data for the nuclear power plant operation. The greatest progress has been done in this last item.

Initially, information was stored in two databases, one of them for analogical signals and the other for digital signals. Under such conditions, the operator, besides being capable of observing indications and desk recorders, only received at his monitor records corresponding to:

- Change of state in valves.
- Start up of components.
- Location of the variables in an alarm or trip zone.

On the other hand, the operator was able to have at his disposal those data corresponding to two months prior to the current date, obtaining them directly from his computer hard disk.

Having at hand the computer collected data and adding local data entered through a special device, two improvements were carried out related to the information presentation:

During 1992 an auxiliary program was introduced enabling the operator to have all the computer information through the creation of visualising screens, showing component graphs and trend diagrams. The system runs under DOS and with VGA resolution.

In 1997 an improvement to this visualising and information integration system was performed, enabling the visualising screen to be presented more sharply, because a high graphic resolution was available.

The above mentioned arguments and the fact that the new system enables to handle four algebraic variable expressions in each visual object of the historical diagram type, together with the extensive participation of the operation staff, enabled an easy and global implementation of the new system.

The system works either under Windows 95 or Windows NT 4.0, thus enabling a friendly interaction between operator and application. The system provides navigation tools enabling a hierarchic screen handling, starting by screens describing process systems, then those corresponding to sub-systems and ending with the display of the whole information available regarding components.

On the other hand, it is possible to incorporate local values, which are included in the data acquisition computer databases.

In connection with the improvement of the alarm-signals display, it must be noticed that its original design does not make any distinction between the different types of alarms or failure warnings; that is the reason for the modification of the colours used in the different kinds of alarms, which makes easier a quick visualisation of them.

In the next scheduled shutdown, the position of the status warnings of some console tables will be changed, in order to put them separated one from another, and several alarms will be modified, being under development a study for the application of an "Off Board" criterion, in which as a first instance, and under normal operation conditions, it will show the alarm board off.

12.4.1.1 **Improvements to the layout distribution**

Concerning layout distribution, a division and enlargement of the control room has been made, making it possible for the shift head to hold meetings without disturbing other people in the control room, and without his physical presence being an interference in it. Moreover, physical barriers have been modified or eliminated, in order to enable a better circulation in that room.

12.4.2 **IMPROVEMENTS TO EMBALSE CONTROL ROOM**

CNE control room has modern design however, some minor changes were made mainly related to:

- General illumination of the room.
- Improvements in the printing data systems.
- Modifications in the access control to the main control room, in order to avoid unauthorised access, and to reduce circulation of people through it.
- Modifications in the alarm systems for those situations leading to a reactor trip.

12.5 COMPLIANCE WITH THE OBLIGATIONS IMPOSED BY THE CONVENTION

Methods to prevent, detect and correct human factors are in an initial stage of development. However, taking into account the experience gathered during the first stage of CNA-I probabilistic safety analysis and the periodic training of CNE staff at Gentilly-II simulator in Canada, as well as the human reliability analyses to be carried out starting in 1998 for the level 1 probabilistic safety analysis of CNE and CNA-I outage, it may be concluded that in the country, some steps have been taken to ensure that the capabilities and limitations of human performance are taken into account.

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

ARTICLE 13

QUALITY ASSURANCE

13.1 INTRODUCTION

The application of proper quality assurance programs in the design, construction, commissioning, operation and decommissioning stages of nuclear installations is a regulatory requirement in the country.

In the case of nuclear power plants, standard AR 3.6.1[1] sets the requirements to be fulfilled. In addition, standard AR 3.7.1 determines when the Responsible Organisation must present the quality assurance program and manual.

Moreover, the operating licenses for CNA-I and CNE set that both must have quality assurance programs during operation. In particular, CNE operating license requires special procedures for modifications in its control computers software, as an additional quality assurance requisite. CNA-II construction license includes a quality assurance program during the construction stage, among other requirements.

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

By means of audits carried out according to the usual methodology (see Section 7.3.3), the Regulatory Body controls nuclear power plant quality programs implemented by the Responsible Organisation.

13.2 QUALITY ASSURANCE PROGRAM OF THE LICENSEE

Since the creation of Nucleoeléctrica Argentina (Company in charge of the nuclear power plants operation), it was considered convenient to dispose of a general quality assurance program which will be the reference frame for the specific quality programs of each organisational unit.

During 1996 and 1997, work was done on the adaptation of the new quality concepts and principles distributed by the International Atomic Energy Agency.

This activity leads to the development of a General Quality Assurance Program that considers the requisites established by the Regulatory Body and the requirements set by the International Atomic Energy Agency in its 50-C-Q "Code

[1] This standard, reviewed in the mid 1997, is consistent with the Code 50-C-Q of the IAEA's NUSS program. The previous version of this standard, named "Quality Assurance", was consistent with the previous NUSS Code 50-C-QA

on Quality Assurance for Safety in Nuclear Plants and other Nuclear Installations" and other Quality Assurance guides.

The program is described in the General Quality Assurance Manual, in general procedures and in specific manuals with their corresponding procedures and instructions.

The General Manual, approved and set in force in November 1997, contains a declaration of the quality policy of the company and organisation, and establishes the basic applicable quality requirements.

General procedures regulate activities related to the different organisational units (see Sections 11.2.1 and 11.2.2), establishing responsibilities and interfaces.

Specific manuals, procedures and instructions describe quality programs of the company organisational units having operation or construction licenses, or carrying out complex activities related with nuclear safety or availability of the nuclear power plants stand-by systems, or performing projects in which nuclear safety is involved. The quality assurance program for each nuclear power plant is described in the respective manuals:

- Quality Assurance Manual for CNA-I Operation (MGC-0).
- Quality Assurance Manual for CNE Operation (MGC-0).

Since 1997, when the General Manual was set in force, the different organisational units have begun adapting their specific manuals and procedures according to the new requirements.

Table 13.1 shows the status of the Operation Organisation General Quality Assurance Program, updated to July 1998.

Not only the General Program but also the specific programs are systematically evaluated in order to ensure their implementation and fulfilment. This task is developed within the company in the following way:

- Quality Assurance Divisions of each organisational unit, are responsible for carrying out the evaluations of specific quality assurance programs of such units (see Figures 11.2 and 11.3).
- Quality Assurance Department of the company is responsible for the execution of assessments of Specific Programs and the General Program (see Figure 11.1).

Table 13.1 - Status of the Quality Assurance Program

ORGANISATIONAL UNIT	DOCUMENT	REVISION	NUMBER OF PROCEDURES
NASA	General Manual	Rev. 0 Updated	General Procedure 10
CNA-I	Quality Assurance for Operation Manual	Rev.1 Under revision	200
CNE	Quality Assurance for Operation Manual	Rev. 3 Under revision	460
CNA-II	Quality Assurance for Construction Manual	Rev.1 Under revision	60
Engineering and Services	Quality Assurance Manual of the Services Dept.	Rev.4 Updated	150
SEU(*) Project	Quality Assurance Manual for SEU	Rev .2 Under revision	8

(*) Slightly enriched fuel elements (see Section 6.2.1.8)

The programs performed in CNA-I and CNE during the design, construction and commissioning stages showed very different characteristics according to the initiation date for each project; this is the reason why they are described separately. The same comment is valid for CNA-II, at present under construction.

13.2.1 ATUCHA I NUCLEAR POWER PLANT

13.2.1.1 Design

CNA-I's design was initiated at the end of the sixties, when the implementation of formal quality assurance programs was internationally incipient. The expansion of quality assurance criteria was by that time beginning in USA, where they were initially applied to the naval and spatial industry, and then gradually extended to different branches of the industry including the nuclear one.

The quality systems applied in the European industry at that time, particularly in Germany, gave special emphasis to some criteria such as:

- The qualification of personnel responsible for the design, construction, assembling and inspection of components was required according to acknowledged strict standards.

- The continuous follow up, during construction and assembling, by independent specialised organisations, to assure the foreseen quality of components, equipment and systems.
- The electrical, electronic and electromechanical component suppliers qualification supported by a wide industrial experience.
- The technical documentation control, particularly construction documents.
- The component testing and inspection, carried out by independent highly qualified and experienced personnel.
- The calibration of measuring instrumentation, in which the German industry is traditionally well-known.
- The responsible personnel qualification in special processes such as welding and thermal treatment.

On the contrary, no other current quality assurance criteria were applied at that time, such as:

- The use of procedures in several controlled activities, e.g. inspection processes and component assembling.
- The use of quality assurance manuals defining organisation, personnel responsibilities, their training requirements and interfaces between the different working groups involved in the construction.
- The documentation review.
- The systematic performing of audits to every controlled activity in order to verify its effectiveness.
- The control of deviations and the execution of corrective actions through a documented system.

Finally, although the quality assurance practices applied to CNA-I design lacked several formalities and documentation aspects that a modern system uses, it had important elements (as qualification of the involved personnel) which have converted CNA-I into a "quality model". This is evident when observing the present state and performance of the equipment and installation systems after more than two decades of satisfactory safe operation.

Due to these circumstances, neither the Responsible Organisation nor the Regulatory Body had the tools to formally evaluate the quality assurance applied by the designer, and, obviously, the corresponding audits were not carried out either.

13.2.1.2 **Construction**

The independent inspector role was performed by two organisations: *Technischer Üeberwachungs Verein, Baden (TÜV)*, appointed by *Siemens - Kraftwerk Union A.G.* Company and Nuclear Power Plants Safety Control and Inspection (CISIN), on behalf of National Atomic Energy Commission (see Sections 8.2.1.2 and 14.1.1.3). For this case, as well as for design, audits were not carried out either.

13.2.1.3 **Operation**

The quality assurance activities started in a more formal way, as in other countries, at the beginning of the eighties. At first, their goal was quality control rather than quality assurance. Nevertheless, once the corresponding manual was produced in 1986, the activities were progressively oriented towards quality assurance.

In 1983, the Regulatory Body carried out an audit of CNA-I operation quality system, with the co-operation of Gilbert-Commonwealth Company[2] as a consultant. In 1990 IAEA, under request of the Argentine Government, carried out an Independent Safety Revision Mission at CNA-I (see Section 6.2.1.1 and Annex 6). The Responsible Organisation, in turn, promoted a peer review, performed by WANO specialists, and the consequent follow-up mission to verify the compliance with the recommendations of such review. In 1996 an IAEA specific mission (IPERS Mission) was carried out to evaluate CNA-I preliminary probability safety assessment report.

13.2.2 EMBALSE NUCLEAR POWER PLANT

13.2.2.1 **Design**

CNE design began in the seventies. At that time, the Responsible Organisation had access to the whole information related to the conceptual and detailed design. During the first stages of the project, Atomic Energy of Canada Limited, the project industrial and architect designer, established and implemented a quality assurance program compatible with the state of the art at that moment.

13.2.2.2 **Construction**

CNE made up and put into practice a quality assurance program for the commissioning stage, and before its commercial operation and in fulfilment of a Regulatory Body requirement, it developed and implemented an operation quality assurance program.

During the last months of 1980 and the beginning of 1981, the Regulatory Body performed a quality assurance audit of CNE project. This regulatory audit was carried out with the cooperation and advice of Gilbert-Commonwealth Company acting as consultant company.

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[2] Specialized Quality Assurance organisation from USA

13.2.2.3 **Operation**

During the nuclear power plant operation, the Regulatory Body performed a number of audits to the quality assurance program of the Responsible Organisation.

The Responsible Organisation, for its part, carried out its own control through the Quality Assurance Department. During 1995 the Responsible Organisation requested a peer review, carried out by WANO specialists.

In November 1997 an OSART Mission was carried out at CNE nuclear power plant (see Annex 7).

13.2.3 ATUCHA II NUCLEAR POWER PLANT

As it has already been mentioned, the construction license sets that the Responsible Organisation shall carry out the project activities according to the quality assurance program presented in the Preliminary Safety Report. Between 1986 and 1992, the Regulatory Body has carried out specific audits to the quality assurance program, related to its implementation, to the component manufacturing and additionally to the conservation of materials and components at the working site.

13.3 COMPLIANCE WITH THE OBLIGATIONS IMPOSED BY THE CONVENTION

Adequate Quality Assurance programs related to activities important to safety throughout nuclear installations life, have been implemented. Therefore, the country complies with the obligations imposed in Article 13 of the Convention on Nuclear Safety.

ARTICLE 14

ASSESSMENT AND VERIFICATION OF SAFETY

14.1 SAFETY ASSESSMENT

The maintenance of an adequate safety level in nuclear installations is a requirement of the regulatory system. Therefore, since the initial stage of a nuclear power plant project until its decommissioning, the Responsible Organisation performs different studies, either as response to a Regulatory Body requirement or as demand of the Responsible Organisation itself. The Regulatory Body controls such safety level by means of audits and evaluation of the studies carried out by the installation or performing its own analysis.

Safety assessments constitute the basis and technical support of the regulatory control. The need for carrying them out comes either from the inspection and audit results in the case of assessments performed by the Regulatory Body, or from the information emerging from abnormal events or what is learnt through accumulated experience in the case of the assessments carried out by the Responsible Organisation. Information about abnormal events and accumulated experience comes from the nuclear power plants itself and other domestic and foreign installations.

The Regulatory Body personnel, specialised in radiological and nuclear safety, use modern computational tools, carry out analyses and evaluations of the information and documentation supplied by the Responsible Organisation as well as their own studies.

The safety assessments above mentioned involve the periodic revision of the probable failure modes of structures, components and systems, and their consequences.

The Safety Report is an important document containing the development and results of the radiological and nuclear safety studies carried out by the Responsible Organisation. In this respect, the AR 3.7.1 standard sets the rules for the submission of the preliminary and final versions of such report, during the licensing process. Its periodic updating, including all the modifications performed to the installation as well as its safety improvements, is established in the operation licence.

On the other hand, the licensing process begins several years before the nuclear power plant operation. First of all pre-operational studies are carried out aiming at evaluating the interactions between the installation and the environment. Such studies include evaluations of the site's meteorological, geological and hydrological characteristics as well as the human activities in the zone of influence of the installation (see Section 17.5). Its results mainly contribute to identify the initiating events, either natural or man-induced, evaluate the radiological consequences of those accidents postulated in the safety analysis, elaborate an emergency plan and determine discharge limits of liquid and gaseous radioactive effluents of the installation. Such information is then compiled and documented in the preliminary and final safety reports.

Furthermore, probabilistic safety studies and other evaluations are usually carried out in order to analyse the impact on safety caused by modifications in the installations, by the occurrence of significant events or by any other justified reason.

For each of the existing nuclear installations, the safety assessments carried out during the design, construction, commissioning and operation stages, are summarised in what follows.

14.1.1 **ATUCHA I NUCLEAR POWER PLANT**

The design of this nuclear power plant was carried out by the end of the sixties, when the main criteria, for the design basis accidents, were based upon the deterministic conception of safety, particularly in the so called "maximum credible accident" criterion. On the other hand CNA-I was purchased by the National Atomic Energy Commission in a "turn key" style, and for this reason information about conceptual design of the nuclear power plant and accidental scenarios was only partially available. Due to these circumstances, both the Responsible Organisation and the Regulatory Body had to examine other accidental scenarios in order to verify the installation safety.

14.1.1.1 **Siting**

The studies related to CNA-I site (see Section 17.5.1) comprised several aspects such us:

- Geographic location, rural and nearby city population distribution and terrestrial and fluvial communication routes.
- Paraná River draining regime, extreme hydrological phenomena, dissipation capacity as final heat sink, hydrologic dispersion.
- Geotechnical characteristics of the site.
- Industrial and agricultural-cattle-breeding uses on the influence zone.
- Meteorology, extreme meteorological phenomena, atmospheric dispersion.

The information collected through these studies had several applications, for instance:

- Determination of the house-of-pumps elevation above zero level and the discharge to river of the recuperative turbine condenser cooling-water.
- Design data for building construction (wind speed in tornadoes, mechanic characteristics of soil, etc.) and liquid and gaseous radioactive effluent retention and release into the environment systems.
- Determination of the discharge limits of liquid and gaseous radioactive effluents into the environment.
- Implementation of emergency plans.

14.1.1.2 **Design and construction**

In May 31st 1968 a contract between the National Atomic Energy Commission and Siemens - Kraftwerk Union AG was signed for the construction of a nuclear power plant having a reactor loaded with natural uranium fuel and heavy water as coolant and moderator. The National Atomic Energy Commission required a preliminary safety analysis as a condition for the authorisation of the plant construction; it would later on convert into the final safety analysis required in the plant operating license.

It was also established that concerning radiological and nuclear safety, the design should comply with standards, rules and in-force laws of the Federal Republic of Germany. The National Atomic Energy Commission should also have drawings, documentation, entrance to building under construction and equipment and component fabrication plants, as well as the possibility of carrying out those inspections considered as necessary.

The most complete CNA-I safety assessment during design and construction stages is the Safety Report. It mainly includes data related to site characteristics, basic technical design of nuclear and radioprotection systems, liquid and gaseous radioactive releases into the environment as well as the protective actions against credible failures in the installation.

This Safety Report clearly reflects the deterministic criteria in which the nuclear power plant design was supported. However, in chapter 13 of such report, a series of perturbations (accidental sequences) and their corresponding plant responses are analysed enabling to have an opinion about the installation safety. The Safety Report is at present being completely revised.

On the other hand, deterministic assessments have been carried out related to the plant backfitting analysis, such as electric system improvement and second heat sink construction. Besides, some modifications to the design have been performed as a result of the probabilistic safety analysis, giving top priority to the so called "dominant contributors to risk", and enabling the qualitative and quantitative comparison of different design and plant operation mode options. The probabilistic safety analysis results showed some weaknesses both in design and operation; the Regulatory Body required the immediate implementation of the corresponding corrective actions (see Section 6.2.1.6).

14.1.1.3 **Commissioning**

With the purpose of carrying out both safety assessment and independent inspections as it had been established in the before mentioned contract, the National Atomic Energy Commission signed, in 1969, a contract with the German Inspection Organisation *Technischer Üeberwachungs Verein, Baden* (TÜV).

In 1971, the *Technischer Üeberwachungs Verein, Baden* issued a report concerning CNA-I construction, mainly containing a series of requirements, recommendations and additional information requests. It also carried out inspections to the fabrication of electric mechanical components assigned to CNA-I. Later on, and during the electric and mechanical assembling stage, it designed a test and inspection plan for safety related systems. In 1972 the contract with *Technischer Üeberwachungs Verein, Baden*, ceased and the National Atomic Energy Commission held the responsibility of carrying out the test and inspection plan.

A commissioning ad-hoc committee called Nuclear Power Plants Safety Control and Inspection (CISIN) was then constituted within the National Atomic Energy Commission, with the responsibility of evaluating and putting into practice requirements, recommendations and additional information requests still pending, as well as advising their authorities in what concerns CNA-I and its operation personnel licensing process.

14.1.1.4 **Operation**

Safety assessments concerning operation cover all the plant operating modes and include a periodic revision of failure modes of structures, systems and components, identifying the consequences of such failures as well. As the plant has been operating for about 24 years, and some original operation safety criteria were different from those used nowadays, it is necessary to make an additional effort in order to take into account the application of new safety criteria.

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

The probabilistic safety assessment carried out in CNA-I, is the most significant safety assessment performed up to now in this nuclear power plant.

The Regulatory Body required the Responsible Organisation to perform a probabilistic safety assessment due to:

- The modifications introduced in systems and operative procedures of the installation, particularly the power increase and those related to the R06 channel failure (see Section 6.2.1.1), required a revision.
- The state of the art at that time enabled the elaboration of a safety study on a probabilistic basis, which would not have been possible in the seventies, during the design and construction stages and the first years of operation of the installation, when there was an insufficient development of such analysis tool.

This probabilistic safety assessment, that includes loss of coolant events classified in 5 groups, 17 plant transients and other initiating events (for instance the pressure vessel failure), made it possible to select the set of accidental sequences mainly contributing to core damage. During its performance several maintenance procedures and periodic tests were analysed and the thermohydraulic analysis of the main accidental sequences was re-evaluated.

A probabilistic safety analysis is now being carried out for CNA-I shutdown state.

It should be mentioned that CNA-I probabilistic safety assessment was reviewed by an IAEA IPERS mission, whose opinion was favourable. All the IPERS recommendations were taken into account.

Other safety studies performed by the Responsible Organisation are:

- CNA-I probabilistic safety evaluation, with the purpose of demonstrating the compliance of the plant with AR 3.1.3 standard.
- Updating of the emergency cooling system (CNA-I Backfitting Study Second Heat Sink - June 1996). The evaluation included a series of thermohydraulic accident analyses with a small loss of coolant, carried out by Siemens - Kraftwerk Union AG. The analyses showed some deficiencies in the emergency core cooling system. These results led to the development of the second heat sink project (an alternative system, independent from the existing one, for the extraction of the core decay heat - see Section 6.2.1.4).
- Updating of the emergency power supply system (CNA-I Backfitting Study Emergency Power Supply - June 1996). The results demonstrated the need for improving the physical independence for the emergency power supply system (see Section 6.2.1.5).
- Evaluation of the reactor internal components after the R06 channel failure.
- Development of a Policies and Principles Manual.

14.1.2 EMBALSE NUCLEAR POWER PLANT

14.1.2.1 Siting

Since 1974, i.e. ten years before the nuclear power plant commissioning, the data about the environmental characteristics at the installation site were collected (see Section 17.5.2). Studies for determining the hydrological aspects of Rio Tercero river basin and particularly of Embalse Lake were performed.

The hydrological characteristics and the use of water for public consumption and irrigation were determined. Concerning this aspect not only the use of hydrological resources was determined but also projections were made contemplating the growing consumption trends due to the expected economical development of the area.

Another aspect taken into account in the pre-operational studies of CNE, was the meteorological condition of the zone, with data in the period 1972-1981, obtained from the micro-meteorological station in operation since 1971 at the site.

A climatological study was carried out to evaluate the average behaviour of daily and annual environment temperature, the variation of lake water temperature as a function of depth, relative environmental humidity, direction and speed of wind, persistence of extreme temperature and humidity values, the annual behaviour of rain and its relationship with wind direction.

Geological and seismological studies related to site safety were also carried out. Information about site seismology was compiled and seismic risk assessment methodologies were applied. The results of CNE pre-operational studies constitute the "Environmental Report of the CNE Siting and Influence Zone". Several national institutes, including the Universities of Córdoba, Río Cuarto and La Plata, performed biological and ecological studies to evaluate the influence of the nuclear power plant in the environment, for example the condenser cooling-water discharges on the species present in the ecosystem.

14.1.2.2 Design and construction

During the nuclear power plant design and construction stages, the Regulatory Body carried out several safety evaluations based on the information supplied by the Responsible Organisation in the following documents:

- Safety Preliminary Report.
- Design manuals.
- Design guides.
- Drawings, schemes, etc.

CNE design began at the middle of the seventies. Although at that time the safety probabilistic criteria had not reached the present development level, it is accepted that CANDU 600 reactor safety is based, at least in part, on probabilistic approaches. The application of the Safety Design Matrix developed by Atomic Energy of Canada Limited for the revision of this type of reactor proves this assertion. Matrices applied to equivalent Canadian plants enabled the detection and correction of some design weaknesses and the elaboration of operation procedures for abnormal situations, taking into account their limitations.

Nevertheless evaluations of deterministic as well as probabilistic aspects regarding safety were carried out. Concerning the first type, a study was made to evaluate the way in which the design of the nuclear power plant systems, particularly those related to safety, complied with the applicable standards (those of the Argentine Regulatory Body, the designer, Atomic Energy of Canada Limited and the Atomic Energy Control Board standards). Concerning probabilistic aspects, reliability studies of the reactor shutdown system, emergency core cooling system and the dousing systems were analysed. Such studies were carried out by the designer to be used as a means of controlling their availability during operation.

The Regulatory Body created a special committee to co-ordinate the tasks related to CNE licensing (see Section 14.1.2.3). This committee performed its activities during the nuclear power plant construction and commissioning and its main functions were: to analyse the Safety Report, the Commissioning Program, the Quality Assurance Program and other related documents; to perform or request a safety analysis, to carry out inspections and audits and to issue requirements to the Responsible Organisation.

The committee performed, on its own or using the services of third parties, several safety assessments during the construction and commissioning. The seismic re-evaluation of the installation should be mentioned as a significant one.

The design basis earthquake for CNE site was defined by means of an horizontal peak ground acceleration value equivalent to 0.15 *g*. This value was used to design the reactor and services building structures as well as the nuclear process systems, components, safety systems, and cooling and post-seismic power supply systems. Once the nuclear power plant construction began, important additional information was obtained (through satellite data) about the geological and (from the analysis of the consequences of the earthquake at Caucete, San Juan, in 1978) seismic characteristics of the site, which led to a deep revision of the original assumptions for the seismic design of the installation.

The Regulatory Body decided to perform a re-evaluation of the site seismologic characteristics and to review the basis for the plant seismic design. The study was assigned to the *Instituto Nacional de Previsión Sísmica* (INPRES -The National

Authority Regarding Seismic Matters) and to the *Departamento de Estructuras de la Universidad Nacional de Córdoba* (Structures Department of the National University of Córdoba). This last institution sub-contracted the evaluation of the electromechanical components to the "Structural Mechanics Associates" (SMA) an American organisation located in USA.

The seismologic studies carried out showed the convenience of considering horizontal peak ground acceleration values between 1.5 to 2 times the original value, for the verification of the seismic resistance of structures and systems above mentioned. The results of such verification were the following:

The reactor building structures and most of the services building, the safety and cooling systems and the post-seismic power supply system, as well as the components of the primary barrier, satisfied the new calculation conditions acceptably. Nevertheless, it was necessary to reinforce some components, such as the steam generators support columns and the structure of the dousing system, anchorage of ventilation ducts and fans or dampers in some valves and tensors in the heat exchanger of the shutdown cooling system.

In the case of the service building, the level of damage due to the new seismic loads was considered acceptable taking into account the safety requirement of this type of building.

In 1991 the Responsible Organisation carried out an important modification of the original design, that led to the construction of the dry storage of irradiated fuel elements facility. Such facility constitutes the dry storage of irradiated fuel elements system, including: transfer cell, the silo field way of access, silos, baskets for fuel elements, tilting table, special tools for fuel management, transfer flask, etc. The seismic requirements comply with AR 3.10.1 Standard and the geological and seismological studies related to site safety mentioned in Section 14.1.2.1 were used (see Section 6.2.2.4).

14.1.2.3 **Commissioning**

The purpose of the Regulatory Body at the commissioning stage was to verify that the construction and assembling of structures, systems and components of the plant were carried out according to the corresponding design. Besides, the Regulatory Body followed the initial functioning tests of components, equipment and systems, with the purpose of verifying that they comply with their design requirements.

In order to reach these purposes, the Regulatory Body appointed an ad hoc committee called Executive Committee for CNE Licensing and decided the permanent attendance of resident inspectors to the plant, to follow its commissioning.

The Regulatory Body also required the Responsible Organisation to appoint an ad hoc committee, capable of taking decisions concerning the nuclear commissioning, constituted by qualified personnel with well known experience in design, construction and operation of reactors. The members of such committee were appointed by the Responsible Organisation and accepted by the Regulatory Body.

The Responsible Organisation implemented a commissioning program, which provided the basis for initial tests program and related activities, including personnel and equipment availability. The commissioning program was accepted by the Regulatory Body.

The most significant safety assessment carried out during this stage was the revision of the Safety Report final version. Most of the material added as result of the revision was related to the thermohydraulic aspects of different accidental sequences postulated in the safety analysis.

On the other hand, since the project design stage had not have been planned according to an established quality assurance program and some design weaknesses, found during the commissioning stage, implied contradictions with some statements in the Safety Report, the Regulatory Body required to the Responsible Organisation:

1. The performance of a safety assessment of those systems related to water service, power supply and shutdown cooling as well as the plant behaviour during a small LOCA or with the moderator system as core coolant. These requirements are part of the CALIN 122/84 Document commented in Section 6.2.2.5.
2. The revision of the Emergency Operation Procedures based on the safety design matrix and/or the analysis results indicated in point 1 above.

Those evaluations enabled a better knowledge of the plant behaviour in case of failure occurrences and the elaboration of more efficient operation procedures, improving diagnosis capability, operator training and management of such situations.

14.1.2.4 **Operation**

During operation, different safety assessments were and are carried out permanently. The following should be mentioned:

- Analysis of occurred significant events.
- Probabilistic study of the electrical grid (performed by the Responsible Organisation).

- Studies related to the "Dry storage of irradiated fuel elements" project.
- Systematic review of the Safety Report.
- Preliminary analysis of the connection between the two primary circuits for different accidental sequences.

Besides, a level I plant specific safety probabilistic assessment is being carried out (See Section 6.2.2.6).

14.1.3 **ATUCHA II NUCLEAR POWER PLANT**

The site evaluation for this nuclear power plant was almost totally based upon the studies corresponding to CNA-I, due to the similar design characteristics and the proximity of both plants (see Section 17.5.3). Nevertheless, and as a Regulatory Body requirement, the pump house elevation above zero level was reviewed, because after CNA-I commissioning, Yaciretá hydroelectric dam was built upstream the river where both plants are located. Thus, the simulation models representing the drain regime of Paraná River and the extreme hydrological basin events had to be modified to include the hypothetical failure of the wall dam. As a result of these studies, the elevation of CNA-II pump house was increased as compared to that of CNA-I.

Although this nuclear power plant is at present under construction, the evaluation performed shows that the risk imposed by it to the population does not exceed the values considered acceptable by the Regulatory Body in the AR 3.1.3 standard.

From the site related point of view, the design of the CNA-II nuclear power plant included consideration of an explosion pressure wave, as well as extreme meteorological phenomena as tornadoes (see Section 17.5.3).

14.2 **SAFETY VERIFICATION PROGRAMS**

Preventive maintenance of CNA-I and CNE, regular availability tests of stand-by systems, in-service inspection of main components, quality assurance and quality control programs have been established.

The preventive maintenance program is conceived to maintain the state of components, equipment and systems and their operation as originally designed. Periodic tests verify the reliability design values of safety related systems, while the in-service inspections enable to detect any material degradation of main components due to ageing or radiation effects enabling the application of preventive and corrective actions.

The Regulatory Body controls the compliance with the mentioned programs, according to what is established in standards and other documents and regulatory requirements. For this purpose it has a program of inspections, audits and specific safety assessments, the general guidelines of which are detailed as follows.

14.2.1 INSPECTIONS, AUDITS AND SAFETY ASSESSMENTS

The purpose of inspections is to verify the compliance with standards and other regulatory requirements. The Regulatory Body performs its own activities to control safety in a nuclear power plant. The regulatory inspections constitute an important basis for the Regulatory Body to make decisions. The inspection programs use a diversity of methods that may be grouped in the following items:

- *Verification of procedures, records and documentation:*

The Responsible Organisation shall carefully document its activities. Among these documents, an essential basis for the regulatory control, the following may be mentioned: test procedures, quality assurance records, test results, maintenance and operation records and records of deficiencies and abnormal events. In some cases the Regulatory Body analyses these documents as a preliminary activity prior to an inspection.

- *Surveillance:*

The Regulatory Body inspection program foresees the direct surveillance of certain structures, systems, components, tests or activities of regulatory concern.

- *Interviews with personnel:*

Usually the inspectors have direct communication with supervisors and with the personnel who work in safety related activities. Particularly, when some interesting event happens, this communication is imperative to perform its reconstruction and to evaluate the personnel's response.

- *Tests and measurements:*

This technique consists in obtaining data or measurements directly. It is generally used in the radiological protection area.

The outstanding characteristics of the regulatory inspections, be them routine or special (also known as non routine inspections), are presented as follows (see Section 7.3.3).

14.2.1.1 **Routine inspections**

The routine inspections, basically carried out by the resident inspectors representing the Regulatory Body in the installations, are focused on the follow-up of the normal plant activities, particularly the monitoring of processes and verification of the compliance with mandatory documentation.

As mentioned in Section 8.2, the four resident inspectors the Regulatory Body has in the nuclear power plants not only carry out the general inspection of activities having regulatory interest continuously, but also permit a direct contact with the installation personnel and interact with the analysis and evaluation groups.

The routine inspections to the nuclear power plants in operations include Operation, Engineering and Radiological Protection areas.

In the special case of CNA-II the routine inspections comprise:

- Control of storage and component conservation conditions.
- Control of maintenance and test execution tasks of equipment and installed systems.

14.2.1.2 **Special inspections**

The special (or non routine) inspections, in which the Regulatory Body specialists in different subjects participate, are carried out when specific situations are produced or when it is necessary to increase the inspection activities. Typical cases are the programmed and non-programmed outages (incidents).

14.2.1.3 **Regulatory audits**

The regulatory audits are performed to specific sections of the organisation that perform maintenance activities, quality assurance, periodic tests, radiological protection systems, and other systems of regulatory concern. Their purpose is to evaluate in an exhaustive way the quality of the tasks performed, according to the provisions in the mandatory documentation. The results produce requirements, recommendations and additional information requests to the nuclear power plant Responsible Organisation.

In general, the purpose of the regulatory audits is to examine the degree of compliance with the provisions in the mandatory documentation. They are planned, controlled, co-ordinated and executed to cover organisational, operative or process aspects of the nuclear power plant and they are carried out by a team of specialised personnel of the Regulatory Body.

At the end of the audit, the group produces a report containing: audit purpose and scope, listing of applicable and reference documentation, constitution of the audit group, plant personnel interviewed, summary of results, conclusions and recommendations. Implementation of the mentioned recommendations is verified through follow up audits.

It should be mentioned that regulatory audits have been carried out on:

a) The periodic test system of both nuclear power plants. These audits are part of the control program of the periodic tests applied in them.
b) The predictive and preventive maintenance activities of CNA-I and CNE.
c) CNA-I radiological protection system.
d) CNE waste management system.

As a consequence of the audit conclusions, some requirements and recommendations were issued the Responsible Organisation. The program mentioned in point a) above includes the follow-up of the compliance with the requirements and recommendations issued.

14.2.1.4 **Safety assessments**

Two complementary methods are mainly applied in the evaluations: the deterministic and the probabilistic one (see Sections 7.2.2, 7.2.2.1 and 7.2.2.2).

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

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

14.3 MAINTENANCE

14.3.1 INTRODUCTION

Maintenance tasks are divided into preventive and corrective. For preventive maintenance tasks, a program is available in which frequency and scope of each equipment or component maintenance is indicated. For early detection of failures maintenance is complemented with techniques such as vibration analysis, ultrasound, eddy currents, infrared analyses, etc.

Preventive maintenance is carried out being the plant in operation and during programmed outages. During these last ones, the big components that must be checked only during shutdown are controlled, such as the main pumps of the primary circuit, moderator pumps, etc.

Corrective maintenance is carried out every day, according to the priority assigned to the solution of failures.

Every task is supported by a working plan containing indications for its performance, including ALARA principle considerations.

For the case of maintenance tasks never carried out before, it is required that a working plan be written and approved before its performance, which must be done in collaboration with Engineering, Operation, Maintenance and Radiological Protection sections.

Each equipment has its historical report containing information related with its time of use, failures and preventive and corrective maintenance actions performed on it. This equipment historical report is used as a basis for the improvement of both preventive and corrective maintenance, being also used as database for the determination of spare parts stock and failure frequency needed for the probabilistic safety assessment.

The programmed outages frequency for nuclear power plants maintenance was determined as a function of the following permanent criteria:

- Fulfilment of in-service inspection program.
- Revision of big components according to the manufacturer recommendations.
- Periodic tests of safety systems, which require that the plant be out of service for their execution.
- Steam generators inspection.
- Optimisation of occupational exposure.
- Optimisation of contractors services.

Since the beginning of the nineties, the following tasks were added to those already considered in the permanent criteria during the programmed outages:

Atucha I Nuclear Power Plant

- Replacement of fuel channels covered with Stellite-6 alloy by others fuel channels containing an LC-1c alloy without cobalt (see Section 6.2.1.3).
- Backfitting: Second heat sink (See Section 6.2.1.4) and emergency power supply system modification (see Section 6.2.1.5).

Embalse Nuclear Power Plant

- Pressure tube inspection program (see Section 6.2.2.2).
- Feeders inspection as a part of the in service inspection program.

14.3.2 AGEING

Both nuclear power plants have inspection programs for their components (Section 6.3.1.3). Such programs comprise all the components safety related.

The inspection of steam generators for both nuclear power plants, CNA-I reactor pressure vessel and CNE pressure channels represent an important part of the above mentioned programs. Due to the fact that these components determine the plant lifetime, the knowledge of the effects of ageing enable to take decisions related to the lifetime management of the installation.

For the case of steam generators, the tube wall thickness is controlled by means of the eddy current technique, as well as the crud deposit on the tube sheet, the steam generator fouling and the effect of impairment due to vibrations.
These inspections generate different actions:

- Plugging tubes with a thickness decrease greater than or equal to 50%.
- Cleaning of crud by means of water jet techniques.
- Chemical cleaning.
- Anti-vibration devices location.

The steam generators present situation is shown in Table 14.1.

Table 14.1.- *Steam Generators present situation*

NUCLEAR POWER PLANT	Number of Tubes	Number of tubes plugged	% of tubes plugged
CNA-I	7938	138	1.7
CNE	14200	27	0.2

The steam generators status is considered satisfactory, so that their replacement is not foreseen before the end of the nuclear power plant lifetime.

CNA-I pressure vessel is referred to in Section 6.2.1.7.

CNA-I moderator heat exchangers were designed in such a way that their inspection is not foreseen. Besides, due to the fact that there are lower temperatures in the moderator circuit, they have accumulated fission and corrosion activated products and therefore they are among the components having a highest level of irradiation in the whole plant.

Their cleaning and inspection will not be carried out until the replacement of coolant channels (being the main contributors to cobalt production) is completed. Nevertheless, a specific group has been constituted in order to act if tube plugging is required. To that purpose, a set of moving shielding, cutting, welding inspection and plugging tools have been designed to facilitate the task if needed. It has been estimated that with the help of such tools, an intervention in one of the moderator heat exchangers could demand some 20 days with an approximate collective dose of 2 man Sv

The follow up of instrumentation and control components deserve especial attention. Specialists belonging to the engineering management section of the Responsible Organisation together with the engineering sections of both plants, have undertaken the supervision of electronic and instrumentation components potentially affected by ageing or obsolescence (see Section 6.3.2).

The components having been discontinued in their manufacturing by the original vendor constitute a special case.

The assessment of these three aspects (ageing, obsolescence and manufacture discontinuation) may be globally named as ageing, and requires the analysis of each component which should provide both an answer and a solution to the following items:

- Design specifications and requirements to be fulfilled.
- Reliability demands.
- Classification according to environmental conditions, seismicity, etc.
- Certification of test and simulations.
- Quality Assurance Program.
- Supplier's qualification.

The effects of obsolescence are basically evaluated when a component is to be replaced by another for corrective maintenance purposes. This is generally applied to instrumentation and control components, which are usually modern and easily replaceable items. Normally, changes introduced as a result of obsolescence are associated with corrective maintenance policies (For example: the mercury wetted relays used in safety systems).

The procedure described is also applicable to electromechanical components, particularly the safety related ones.

Up to now the analysis carried out involved those components which revealed ageing features. Recently, the Regulatory Body requested the Licensee to elaborate an ageing management program with the purpose of selecting the significant components on which the impact of ageing should be assessed, analysing ageing mechanisms, estimating of ageing consequences on reliability, calculating of remaining lifetime, determining the method for their monitoring and, finally, suggesting the necessary actions to mitigate the effects of ageing (maintenance, operation and design improvements).

The Regulatory Body also raises issues to be discussed in advance, as in the case of other issues resulting from operational experience (core neutron detectors change at CNE).

CNE pressure tubes are periodically inspected and it is expected that they will operate till the end of their lifetime.

14.3.3 SPARE PARTS STOCK OF THE NUCLEAR POWER PLANTS

Both CNA-I and CNE nuclear power plants began their commercial operation with a complete spare parts stock. This gave enough time for both nuclear power plants to organise stock strategies and to determine the minimum stock value for critical spare parts. The priority criteria and other factors taken into account for this task were as follow:

Priority criteria:

- Spare parts belonging to safety systems have priority. Although the systems are redundant, the failure of any of the redundancies must be repaired in some hours time, as established in the Policies and Principles Manual. When the expiry date is reached, the plant must be shutdown and driven and maintained in a safe state.
- In second place in order of priority, there are those non redundant spare parts or components the failure of which leads to an out of service state, for instance main pump seals, spare parts of the refuelling system, turbine, etc.

Other factors:

- Replacement frequency. It is usually based in the lifetime value given by the manufacturer or in the historic value based on the plant experience or that of any other equivalent installation (for instance any other CANDU 600).
- Time required for the arrangement of prevision. It depends on the place where the spare part is manufactured (domestic or foreign), time of transport and customs management.

- Time of delivery. It is determined by the manufacturer according to the number of spare parts, particularly if the supplier has no stock.

Depending on all these data, a minimum stock value is established taking also into account unforeseen delays during the whole process.

Another way of buying spare parts is determined by the programmed outages, for which a specific purchase is carried out of those spare parts which may be required during preventive maintenance of components and systems.

In most of the cases, the purchase of spare parts is done through the nuclear power plant suppliers: Siemens - Kraftwerk Union AG for CNA-I, Atomic Energy of Canada Limited and Societa Italiani Impianti P.A. for CNE. These make the purchase management easier, because these companies have dealers in the country, which is convenient to solve exceptional situations requiring urgent provision of a particular spare part.

In the case of CNE, a new possibility is open through Candu Owners Group, one of its services being the purchase of spare parts for CANDU plants simultaneously. This modality enables an adequate provision of spare parts.

A special treatment is required for those spare parts which have been discontinued in the fabrication of any of their components. In such cases, either a great quantity of them must be bought, or a substitution process must be initiated after analysing the technical specifications and seeking for a new qualified supplier.

It should be emphasised that since the beginning of the commercial operation of both nuclear power plants, no outages or delays in starting the plants up due to lack of spare parts were produced.

14.4 COMPLIANCE WITH THE OBLIGATIONS IMPOSED BY THE CONVENTION

From the preceding considerations, it becomes evident that in the country periodic and detailed safety assessments are carried out along every stage of the nuclear power plant lifetime. Such assessments verify that the nuclear power plant state and operation conditions correspond to the design previsions, to the limits and conditions imposed by the Regulatory Body and to any other national safety requirement. Therefore, the country complies with the obligations imposed in Article 14 of the Convention on Nuclear Safety.

ARTICLE 15

RADIOLOGICAL PROTECTION

15.1 GENERAL CRITERIA

The radiological protection basic criteria applied in the country establish that:

- Practices using ionising radiation shall be justified.
- Radiological protection shall be optimised.
- Limits and established dose restrictions shall not be exceeded.
- Accidents shall be properly prevented and mitigate their radiological consequences if they occurred.

These criteria have been applied in the country for more than two decades.

The justification criterion sets that any practice that implies, or could imply, personnel exposure to ionising radiation will only be justified if it originates a net positive benefit to the society. In the case of installations involved in this report it is not considered necessary to give more details for its justification.

As regards the optimisation of radiological protection systems, it is the policy of the Regulatory Body to require that personnel exposure due to a justified practice be kept as low as reasonably possible, taking into account social and economical factors (ALARA). In order to achieve optimisation, the Regulatory Body requires that the technically available options and the collective dose reduction be detailed as well as the cost associated to each option.

The effective dose limit for members of the public is 1 mSv in a year and it is applied to the average total effective dose to the critical group due to all installations and practices. The annual equivalent dose limits are 15 mSv for the lens of the eye and 50 mSv for the skin.

In order to consider the contribution to the dose received by the critical group due to practices carried out at regional and global levels, and to count on a proper margin for future practices, the Regulatory Body has established dose restrictions (dose constraints) for a particular installation, not only on the effective dose but also on the collective effective dose. For the case of a nuclear power plant, these restrictions are:

- The effective dose to the critical group shall not exceed 0.3 mSv in a year.
- The collective effective dose shall not exceed 15 [man Sv $(GW_{(e)}\ y)^{-1}$] of generated electric energy.

In order to apply these restrictions and not to exceed dose constraints, the Regulatory Body limits the authorised discharges into the environment (discharge constraints) (see Section 15.3). The value of 0.3 mSv is applied when it is demonstrated that the effluent discharge system has been optimised. Otherwise, a more restrictive value is used that shall not exceed 0.1 mSv per installation.

Concerning dose limits for occupational exposure the Regulatory Body establishes that:

- The effective dose limit is 20 mSv per year. This value shall be considered as the average in 5 consecutive years (100 mSv in 5 years), but 50 mSv shall not be exceeded in a single year.
- The equivalent dose limit is 150 mSv in a year for the lens of the eye, and 500 mSv in a year for the skin.

The dose limits apply to the sum of the significant doses from external exposure in the specified period and the significant committed doses from intakes in the same period.

These limits values have come in force in January 1995 and since then, the value of the effective dose integrated over five years is also accounted for.

15.2 STANDARDS RELATED TO RADIOLOGICAL PROTECTION

Previous to the nuclear power plant commissioning, the Responsible Organisation shall submit to the Regulatory Body the mandatory documentation established in standard AR 3.7.1, which mainly includes:

- Safety Report.
- Operating Manual.
- Radiological Code of Practice.
- Maintenance Manual.
- Emergency Plan.
- Monitoring Manual.

On the other hand, the Regulatory Body criteria concerning radiological safety in nuclear power plant have been defined in standards AR 10.1.1, 3.1.1, and 3.1.2.

Apart from the mandatory documentation, the Regulatory Body requires additional documentation clearly demonstrating that the operation of each nuclear power plant is performed within the authorised limits and operating conditions and that enables, at the same time, the evaluation of eventual deviations. In that sense, records containing information about operation, deviations from limits or operating conditions, human errors or safety system failures, etc. are required. (see Sections 9.3, 19.3.1 and 19.4.1).

The Basic Radiological Safety Standard (AR 10.1.1) establishes the general guidelines required to reach a proper level of protection against the harmful effects of ionising radiation and of the radiological safety of the installations or practices involved.

Standard AR 3.1.1 determines the design criteria that shall be satisfied related to the nuclear power plant occupational exposure, establishing the dose constraints during normal operation, maintenance, repairing and inspections of those installations.

Standard AR 3.1.2 refers to the limitation of radioactive effluent discharge to the environment, establishing the total effective dose to the critical group and the collective effective dose constraints.

On the other hand, the mandatory documentation sets that apart from relevant events, and according to the provisions in the operating license, the following radiological protection information shall be communicated to the Regulatory Body in due time and form:

- Discharges of liquid and gaseous effluents into the environment (quarterly and annually).
- Dose absorbed by the personnel (quarterly and annually).

Regarding potential exposures, the Regulatory Body has developed a probabilistic criterion with the purpose of limiting individual risk in members of the public. For each installation, the individual risk associated to a given accidental sequence should have at most the same value as that associated to normal situations at such installation. This design criterion is contemplated in standard AR 3.1.3 and has been applied in Argentina during the last 15 years.

The AR 3.1.3 standard establishes that:

- Every reasonable measure that could contribute to avoid accidents shall be taken to minimise the associated radiological risks.
- It is necessary to identify every accidental sequence, which in case of occurrence could produce unwanted exposure of people to radiation.
- The annual occurrence probability for each of them should be calculated as well as the resulting radiological consequences on the critical group.
- The accidental sequence identification and the annual occurrence probability calculation must be done applying acceptable tools, such as the fault and event tree technique.

15.3 CONDITIONS FOR RADIOACTIVE MATERIAL RELEASE

According to regulatory standards, the radioactive effluent retention systems shall be optimised. The different alternatives considered for effluent treatment should be satisfactorily detailed to the Regulatory Body, as well as the costs of each alternative and the collective effective dose reduction achieved in each case. The selection of the best option is carried out according to usual procedures.

When the optimisation is performed by means of a cost-benefit analysis, a value of the proportionality coefficient between the social cost and the collective dose of 10.000 USD per man Sievert is used.

The dose constraints to the population for a particular practice are consistent with those proposed by IAEA, but they are more conservative because of the condition applied both to the individual and to the collective dose (see Section 15.1).

The operating licenses issued by the Regulatory Body for nuclear power plants establishes that the dose to the critical group due to the discharge of radioactive effluents to the environment should be as low as reasonably possible and shall not exceed the constraint given in terms of the following expression:

$$\sum_i \frac{A_i}{K_i} < L$$

where

A_i is nuclide *i* activity released to the environment in the period considered
K_i is a constant activity value, stipulated for the nuclide *i*, for a given installation
L is the limit for this sum of fractions, with different values for the different periods considered; $L=10^{-2}$ in a day, $L=3\times10^{-1}$ in three months and L=1 in a year.

The value of K_i is calculated for each installation, radionuclide and type of discharge (liquid and gaseous) using specific models to estimate the dose to the critical group, taking into account the site characteristics and the critical group location.

This kind of evaluation ensures that if this inequality is satisfied, the dose constraint for people will be not exceeded.

The release of gaseous and liquid effluents occurring during normal operation of nuclear installations is continuously monitored and controlled. In case of detecting significant deviations from historical averages or growing annually discharged activity trends, they shall be carefully analysed and justified.

Besides monitoring effluent discharges, the Regulatory Body requires the implementation of an environmental monitoring program in the installation surroundings, including measurement of activity in water samples, sediments, vegetables, fish, milk and every other sample of the surrounding biosphere.

15.4 ENVIRONMENTAL IMPACT

With the purpose of evaluating the environmental impact due to the nuclear power plant operation, several studies were carried out in the site. Some of these studies included data obtained prior to the beginning of the commercial operation and some others were developed during operation.

Some of these studies aimed at comparing the evolution of significant parameters on the environment before and during nuclear power plant operation. Studies of climatologic, hydric and seismologic characteristics of the region, distribution and population characteristics, dwelling, human activities and agricultural-cattle breeding characteristics as well as eating habits in the zone, should be mentioned.

Besides, dilution factors were calculated in order to evaluate the theoretical radionuclide distribution in the environmental compartments of the man nutrition chain. Moreover, radio-ecological evaluations were performed on vegetable specimens, wild animals, sediments and other components of the ecosystem.

15.4.1 ATUCHA I NUCLEAR POWER PLANT

15.4.1.1 Radioactive release into the environment

The Regulatory Body authorised a set of gaseous and liquid discharges limits, contained in the plant operating license and shown in Tables 15.1 and 15.2 respectively. For critical groups doses, these limits were set much lower than 0.3 mSv.

Table 15.1 –Authorised gaseous discharge limits for CNA-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 CNA-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 CNA-I 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, taking into account experimental data obtained during 1983 and 1986.

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

YEAR	I-131 (TBq)	Tritium (TBq)	Aerosols (TBq)	Noble Gases (TBq)	C-14 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}
AVERAGE	4.5×10^{-4}	5.5×10^{2}	2.0×10^{-5}	1.4×10^{2}	4.1×10^{-1}

The liquid radioactive releases to the environment by CNA-I since its start-up until December 1997 are presented in Table 15.4, discriminating between liquid discharges of H-3 and gamma emitters.

The 90% of the total average discharge from CNA-I to the environment corresponded to tritium. Comparing these discharges with the respective annual authorised discharge limit, it is observed that they were less than 10% of such limit.

15.4.1.2 **Public exposure**

The annual average dose to the critical group due to CNA-I operation, during the period 1974-1997, 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 1974-1997 are shown in Table 15.5, discriminated according to the discharge type.

Table 15.4 – Activity released from CNA-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}
AVERAGE	4.8×10^2	1.5×10^{-1}

Table 15.5 - Individual dose to the critical group for CNA-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}
AVERAGE	3.4×10^{-3}	8.7×10^{-4}	4.3×10^{-3}

The effective collective dose normalised per unit of electric energy generated $GW_{(e)}$ y, 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 CNA-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}
AVERAGE	4.0×10^{-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 CNA-I nuclear power plant, for the period 1974-1997, 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 1974-1997. 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 1974-1997, 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 CNA-I initiated its operation before the above mentioned standard was in force.

15.4.2 EMBALSE NUCLEAR POWER PLANT

15.4.2.1 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 groups doses, these limits were set much lower than 0.3 mSv.

Table 15.7. Authorised gaseous discharge limits for CNE

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 CNE

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 CNE 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 CNE to the environment from the beginning up to December 1997 are presented in Table 15.10, discriminating between liquid discharges of H-3 and gamma emitters.

The 40% of the total average discharge from CNE to the environment corresponded to tritium, and 60% to noble gases. Comparing these discharges with their annual authorised discharge limit, they were less than 10% of such limit.

Table 15.9- Activity released from CNE to the environment as gaseous discharges

YEAR	I-131 (TBq)	Tritium (TBq)	Aerosols (TBq)	Noble Gases (TBq)	C-14 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}
AVERAGE	7.2×10^{-4}	6.6×10^{1}	5.6×10^{-3}	3.4×10^{2}	4.6×10^{-1}

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

Table 15.10 – Activity released from CNE 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}
AVERAGE	1.9×10^{2}	5.0×10^{-3}

15.4.2.2 **Public exposure**

The annual critical group dose due to CNE operation during the period 1984-1997, are presented in Table 15.11, discriminated according to discharge type. The annual average critical group doses due to CNE operation, for the period 1984-1997, resulted lower than 2% of the established individual dose constraint. The liquid discharges were the main contributor.

Table 15.11- Individual dose to the critical group for CNE

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}
AVERAGE	2.0×10^{-4}	5.2×10^{-3}	5.4×10^{-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 CNE

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}
AVERAGE	1.1×10^{-2}	2.2×10^{-1}	2.3×10^{-1}

The average collective effective dose per unit of electric energy generated, calculated with population data up to a radius of 2000 km from CNA-I nuclear power plant, for the period 1984-1997, 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 0.1 [man Sv $(GW_{(e)}\ y)^{-1}$] for tritium in the period 1984-1997 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 1984-1997, 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 CNE's design was finished before the above mentioned standard was in force.

15.5 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.

15.5.1 DOSE LIMITS TO WORKERS

In practice and 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 individual 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 individual 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.

15.5.1.1 **Occupational dose in Atucha I nuclear power plant**

In CNA-I, Co-60 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 CNA-I (see Section 6.2.1.3) and in CNA-II, 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 CNA-I during the period 1974-1997, are presented in Table 15.13.

15.5.1.2 **Occupational dose in Embalse nuclear power plant**

The collective effective dose, the normalised collective effective dose and the average effective dose received by CNE workers during the period 1983-1997 are presented in Table 15.14.

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

Table 15.13 - Occupational Dose in CNA-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	6
AVERAGE	7.8	32	13

Table 15.14 - Occupational Dose in CNE

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
AVERAGE	1.9	3.6	3.3

15.6 REGULATORY CONTROL ACTIVITIES

The control and surveillance of the compliance with standards and other regulatory documents, is completed with a program of routine and non routine audits and inspections, 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, or 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 Section 16.7).

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 (CNEA)

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 CNA-I and CNE 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.

15.7 COMPLIANCE WITH THE OBLIGATIONS IMPOSED BY THE CONVENTION

In the country the necessary measures were implemented in order to achieve an occupational and public exposure to radiation produced by the nuclear power plant operation, which does not exceed the corresponding limits and dose constraints, and is as small as reasonably possible.

Therefore, the country complies with the obligations imposed in Chapter 15 of the Convention on Nuclear Safety.

ARTICLE 16

EMERGENCY PREPAREDNESS

16.1 INTRODUCTION

The Regulatory Body requires from the Responsible Organisation, a plan to respond in case of an radiological emergency not only inside but also outside the nuclear power plant. Such plan, usually known as "Internal and External Emergency Plan", shall comprise every aspect related to the strategy required to control and limit the accident consequences. This plan is part of the documentation required by the Regulatory Body to license the nuclear power plant operation, which shall not be initiated before the plan is accepted.

The execution of the protective measures of automatic application is based on the information coming from the installation itself and the available time for its implementation is usually short.

Due to this, it is the responsibility of the nuclear power plant manager, who becomes the responsible for the emergency, to execute the urgent actions contemplated in the emergency plan (called protective measures of automatic application outside the installation).

With the purpose of being able to assume such responsibility, the Responsible Organisation shall carry out the previous agreements so that the Civil Defence staff, Security Bodies and other involved organisations asked to act during the first instances of the emergency, may report to the nuclear power plant manager.

Once the initial period has elapsed and the established conditions of the emergency plan have been achieved, the responsibility for the off-site actions is transferred to the Civil Defence Chief.

It is also the Responsible Organisation's responsibility to foresee that the required human, material and economic resources are permanently available in order to ensure enough operational capability to face emergency situations.

On the other hand and as an explicit demand contained in the operating license, emergency exercises are annually carried out inside and outside the installation, aiming at the evaluation of the participating groups response and the improvement of the plan. The specific objectives of each annual exercise must be agreed with the Regulatory Body.

16.2 LAWS, REGULATIONS, AND NATIONAL REQUIREMENTS CONCERNING EMERGENCY PLANS

The Regulatory Body has ruled the planning and preparedness of responses to emergency situations in the nuclear power plants, through different documents, e.g. regulatory standards AR 10.1.1 and AR 3.7.1, operating licenses and requirements to the Responsible Organisation and Primary Responsible of the installations (Requirements No 192 and No 222). The requirements to be fulfilled by the nuclear power plant emergency plans are, in general, the following:

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

The urgent protective measures carried out on the basis of the plant situation and meteorological conditions without waiting for radioactive measurements in the environment, are basically the following (see Table 16.1):

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

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

The non urgent protective measures established in the emergency plan are carried out by the Civil Defence staff and pertinent authorities, advised by the Regulatory Body in those aspects of its competence.

In order to carry out this advice, the Regulatory Body has an intervention system for radiological emergencies, constituted by a primary intervention group and a nuclear safety support group. Both groups rely on specialists on radiological and nuclear safety as well as on the specific equipment and logistic infrastructure to carry out measurements of radiation levels in the site surroundings and make estimates of the involved dose in order to enable the Civil Defense staff to take decisions.

The implementation of the non urgent protective measures depends mainly on the measurement results of radioactive material spread out to the environment (see Table 16.1). The most significant are the following:

- *Evacuation* from the zones affected by radioactive deposit. This action must be implemented:
  - Compulsorily, in case the radiation level coming from material deposited on the land reaches or exceeds 100 mSv, integrated during the first 6 hours after the radioactive emission.
  - Optionally, if the same dose (100 mSv) is integrated during the first 24 hours after the accident.
- *Intervention in relation to food.* The intervention levels adopted by the Regulatory Body for the substitution of contaminated foodstuffs for consumption were obtained from an optimising analysis in which the expected effects of such food consumption and the drawbacks produced by their absence or replacement by non contaminated products were taken into account. Due to the country characteristics, the contaminated food may, in general, be replaced by other products coming from areas not affected by the accident, so that the mentioned intervention levels are significantly lower than those from other countries (see Table 16.2).
- *Decontamination of land.* Due to its high cost the implementation of this action shall be decided on the basis of an analysis for each specific case.

16.3 IMPLEMENTATION OF REGULATIONS CONCERNING EMERGENCY PLANS

The basic aspects mentioned in Section 16.2 were established not only in the operating nuclear power plant internal and external emergency plans, but also in the agreements celebrated with official organisations. These agreements, which are part of the emergency plans, enable the use of networks, communication systems and equipment belonging to safety organisations, as well as their quick transportation and fire fight resources.

As regards agreements with other organisations, an important example is the treatment of injured, contaminated or irradiated persons. The primary treatment for these cases is foreseen in hospitals located in zones near the nuclear power plants. In case more specialized health centres for the assistance of patients were required in the City of Buenos Aires, several special agreements have been celebrated with institutions having hospital infrastructure and well trained personnel for such patients (see Section 8.4).

Table 16.1 - Protective measures and intervention levels

	PROTECTIVE MEASURES	APPLICATION	INTERVENTION LEVELS	COMMENTS
Urgent Protective Measures	Control of access and road block	Always	n.a.(*)	Protective measures carried out on the basis of the plant situation and meteorological conditions.
	Sheltering	Always	n.a. (*)	
	Iodine prophylaxis	Always	n.a. (*)	
Non Urgent Protective Measures	Evacuation	Always	100 mSv	Dose integrated during the first 6 h after the radioactive deposition.
		Optional	100 mSv	Dose integrated during the first 24 h after the radioactive deposition.
	Re-entering evacuated zones.	Always	100 mSv	Dose integrated in one year.
	Land decontamination	Optional	n.a. (*)	This action shall be decided on the basis of a case by case analysis.
	Food ban	Always	Yes	See Table 16.2

(*) Non applicable

Table 16.2 - Intervention levels for foodstuff, in Bq/kg, as result of a generic optimisation analysis

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

Group 1: ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Am, ^{242}Cm, ^{244}Cm, ^{239}Np
Group 2: ^{137}Cs, ^{134}Cs, ^{131}I, ^{89}Sr, ^{90}Sr, ^{95}Zr, ^{103}Ru, ^{106}Ru, ^{140}Ba, ^{144}Ce

CNA-I emergency plan was originally elaborated during the seventies, before its commissioning, and then modified in several opportunities. CNE emergency plan was developed during the early eighties, also before its commissioning. In 1995 both nuclear power plant Primary Responsibles carried out a detailed revision of their respective emergency plans, in fulfilment of Requirements No 192 and No 222 issued by the Regulatory Body. Such revision enabled the adequacy of plans to the new criteria related to interventions in radiological emergencies prevailing at national and international levels.

16.4 ON-SITE AND OFF-SITE NUCLEAR POWER PLANT EMERGENCY PLANS

The nuclear power plant internal and external emergency plans collect the necessary information about planning and management, that the installations have, together with public organisations involved in the emergency, in order to face an accidental situation.

The nuclear power plant emergency plans are prepared in such way that the intervention before an accident takes mainly into account the following objectives:

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

The emergency plans of both CNA-I and CNE include:

Concerning organisation and responsibilities:

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

Concerning the procedures:

- They set the installation conditions in which the nuclear power plant manager shall declare the emergency in its different levels:
  * Internal alert state inside the installation.
  * Alert state outside the site.
  * Internal emergency inside the installation.
  * Emergency outside the site.
- They specify the correspondence between the different emergency levels and the alarm levels of Civil Defence.
- They include the following actions to face an emergency situation:
  * Quick emergency detection.
  * Organisation activation in order to face the emergency situation.
  * Evaluation of the situation.
  * Initiation of protective measures application.
  * End of protective measures application.
  * Remedial measures.
- They establish protocols, and detail the communication systems necessary to manage the emergency.
- They specify the protective measures to be applied, according to the type of accident and its possible evolution, mainly in the following cases:
  * Noble gases release only.
  * Noble gases and volatile elements release.
  * Noble gases, volatile elements and aerosols release.
- They include the way of implementing protective measures and indicate:
  * The circumstances in which protective measures shall be implemented.
  * Who will be in charge of their implementation.
  * The areas in which countermeasures must be implemented.
  * Circumstances and way in which the protective measures cancellation shall be decided.
- They specify the protocols for alert communication, information and instructions to the potentially affected population (broadcasting, television, loudspeakers, alarms, etc.).
- They include the protocols for the control of doses to personnel who act during the emergency and the actions to be taken in case their values exceed the corresponding constraints and prescriptions. As regards radiological protection, it should be noticed that the personnel involved in protective measures application, specially the non urgent ones, is considered as occupationally exposed. For the case of persons acting in the application of urgent protective measures or aiming at saving lives, the Regulatory Body has set the criteria included in AR 10.1.1 standard.

Concerning physical places and equipment

- They set the sheltered (under protection) place for the permanent or temporary personnel who perform activities up to a radius of 3 km around the installation.
- They set the places for personnel concentration to perform a possible evacuation, which shall also be capable of giving eventual sheltering.
- They detail the necessary equipment that must be available for radiological monitoring.
- They set the places inside as well as outside the installation for the operation of the Internal Committee of Emergency Control, and indicate their characteristics.
- They set the availability of physical places and required consumable goods, for the implementation of protective measures, particularly those for people temporary relocation in case of evacuation.
- They set who and where shall inform the massive communication media.

Concerning maintenance of resources

- They have a continuous training program for the nuclear power plant staff and for the external organisations participating in the emergency. The program contemplates aspects related to the plan implementation and to general radiological safety.
- They have a procedure to update general and specific contents of the emergency plan.
- They set a program for calibration and maintenance of equipment and instrumentation assigned to the performance of tasks during the emergency.
- They foresee the performance of an annual exercise of the emergency plan application.

16.5 STRUCTURE OF THE EMERGENCY PLAN AT NATIONAL LEVEL

There is legislation related to passive defence systems at the National level. To this respect Act No 14467, 1958 "Passive Defence" and Act No 17192, 1967 "Civil Defence Service", their complementary laws and their respective regulatory decrees are mentioned. These laws are mainly referred to the actions and activities of non aggressive nature conceived to avoid, cancel or decrease the effects of warlike acts, natural agents or disasters of any other origin that could affect the population and their goods. In the same sense, decrees No 270,1992 and No 1041,1995 refer to life preservation, habitat and possessions of people threatened by calamities of natural or antropogenic origin.

At the national level, the mentioned laws and decrees are implemented in an effective way through the Civil Defence National Organisation, organism under the National Ministry of the Interior.

At provincial and county levels, the Civil Defence Body is structured and organised within the field of the Province Security Secretary or, in a direct way, depending on the provincial Executive Power.

The response capability of the Civil Defence Group to radiological emergencies, has been progressively increased during the last years, according to what has been observed in the yearly simulations. Nevertheless, an improvement of the involved personnel training referred to response to such kind of situations is considered necessary.

There is no plan about a nuclear contingency at the national level; nevertheless, those counties that could be directly affected by a nuclear accident, as for instance the County of Zarate, on which the city of Lima depends, for the case of CNA-I and the County of Embalse, for the case of CNE, have incorporated the hypothesis of "accident in a nuclear power plant with emission of radioactive material and environmental contamination" to their General Emergency Plan.

The system of information to the public about countermeasures in the case of an accident, has been developed during the last few years. Nevertheless, it should be emphasised by campaigns of diffusion to the people potentially involved in such accidents.

16.6 CLASSIFICATION OF EMERGENCY SITUATIONS

The emergency plans for both nuclear power plants set the following levels of action taking into account the potential consequences:

Level 0 (Passive level): it corresponds to a situation which is external to the Plant, of natural or diverse origin, and that due to its importance it may conceivably affect the plant normal operation.

Level I: it corresponds to any abnormal situation in the Plant, with consequences of such characteristics as to justify the assumption that, in principle, the annual dose limits for workers established in the AR 10.1.1 standard shall not be exceeded.

Level II: it corresponds to any abnormal situation of such type that, without exposing the public improperly, could produce on the nuclear power plant workers an exposure to radiation with doses exceeding the annual limits established in the above mentioned standard.

Level III: it corresponds to any abnormal situation that could cause an exposure to the public higher than the limits established in that standard.

Each of the described intervention levels corresponds to different alarm levels to be communicated to the Civil Defence staff.

16.7 **EMERGENCY EXERCISES**

The execution of emergency exercises is a general regulatory requirement. In the particular case of a nuclear power plant, the Regulatory Body has set that such exercises shall be performed once a year covering the internal and external aspects of the installation.

The Emergency Plan application exercises are annually programmed and designed by the Responsible Organisation. They shall take into account the objectives established by the Regulatory Body and count with its agreement. These exercises include every aspect of the emergency plan. Usually all the organisms involved participate in them. The Regulatory Body performs the necessary surveillance from the regulatory point of view (i.e. verifying plant personnel response) and executes the tasks described in the plan (i.e. advising the Civil Defence Body).

The exercises are carried out in such a way as to enable the verification of the implementation of urgent (of automatic application) protective measures as well as those requiring and having more time for their implementation.

Personnel of the nuclear power plant, of the Responsible Organisation and of different external organisms (such as Civil Defence, Federal and Provincial Police, "Gendarmería Nacional" (Border Police or National Civilian Police), Maritime Prefecture, Firemen Body, Hospitals, etc.), supported by Regulatory Body personnel, take part in the exercises. During the last ten years the population of the cities of Embalse, Villa del Dique, Villa Rumipal and La Cruz, in the case of the CNE, and those people from the city of Lima, close to CNA-I (essentially people who live within a radius of 10 km around the installations), have also participated.

During the exercises, the nuclear power plant staff behaviour and that of the members of organisations participating in the emergency is evaluated on one hand, and on the other, the population surrounding the installation is trained on the actions to be taken in case of a radiological emergency that could affect them. In that sense the diffusion and public information activity is mandatory, mainly performed during the months preceding the exercise.

After the exercises, meetings among all the involved organisations are carried out. In those meetings, the results of the exercises are evaluated with the purpose of producing conclusions leading to the Emergency Plan improvement. Particularly, the population response is analysed with the purpose of improving not only the implementation of specific protective measures involving public but also diffusion activities.

16.8 INTERNATIONAL AGREEMENTS

In December 1986, the Federative Republic of Brazil and the Argentine Republic signed an Argentinean - Brazilian Co-operation Agreement. Annex II to the Protocol 11 of such agreement includes the program of "Prompt Notification and Mutual Assistance in the Event of Nuclear Accidents and Radiological Emergencies".

As part of the agreement, and with the purpose of achieving a better knowledge of the criteria used in both countries at an emergency situation, in 1992 personnel belonging to the Brazilian Regulatory Body took part in an emergency exercise in CNE, in which six Brazilian specialists participated.

In February 1990, the Argentine Republic signed with IAEA the "Convention on Early Notification of a Nuclear Accident" and the "Convention on Assistance in the Case of a Nuclear Accident or Radiological Emergency".

On the other hand, Argentina is a member and a contact point of the "Network for Medical Attention to Overexposed Persons" of the Pan-American Health Office.

16.9 COMPLIANCE WITH THE OBLIGATIONS IMPOSED BY THE CONVENTION

From the preceding considerations, it may be concluded that in the country there are updated plans capable of facing emergencies in nuclear installations. Moreover, such plans establish the actions to be followed not only inside the installations but also outside them, and periodic exercises are carried out for their application. Additional efforts shall be made to establish measures to be imposed to the public concerning emergency preparedness and response capability of support organisations. Nevertheless, in spite of such additional efforts the country complies with the obligations imposed in Chapter 16 of the Convention on Nuclear Safety.

ARTICLE 17

SITING

17.1 INTRODUCTION

The objective of the siting studies is to select a suitable site for a nuclear power plant, including appropriate assessment and definition of the related design bases, taking into account that nuclear power plant design implies the consideration of site dependant factors which may affect, directly or indirectly, the plant safety. For instance, the capability and reliability of the ultimate heat sink and power supply networks, the potential for natural and man induced events, and the characteristics of communication routes and accesses.

Therefore, these siting studies are aimed to determine the effects of external events occurring in the region of the site, to evaluate the potential radiological impact on the environment due to the plant operation and the feasibility of the emergency plans.

In the country, these studies (for selecting the location of a nuclear power plant) are part of the requirements the licensees shall comply with at the time they request a construction license. In the licensing process for a nuclear power plant, a previous and independent licensing of a site is not explicitly required.

However, when the construction license of a plant is required for its subsequent approval by the Regulatory Body, the criteria and procedures followed in performing the siting studies (as well as the results obtained) are reviewed taking into account that they should have complied with the established requirements, and the state-of-the art and recognised practice on this matter at the time.

Summarising, the siting studies for a nuclear power plant construction enable the analysis of the following aspects:

a) The selection of possible sites and the evaluation of their suitability for the construction and operation of such installations.

b) The establishment of safety requisites related to the site conditions.

c) Effects on the installation of the external, natural or man-induced events, that may occur in the site region.

d) Characteristics of the site and its surroundings related with the exposure pathway of radioactive materials released to the environment during the normal operation or as a consequence of accidental situations.

17.2 EXISTING SITES

In the country, two sites were in due time selected and evaluated as suitable for nuclear power plant construction: Atucha, at the right bank of the Paraná de Las Palmas River, in the Province of Buenos Aires, and Embalse, at the coast of the Río Tercero Dam Lake, in the Province of Córdoba.

Atucha site has two independent units, one in operation, CNA-I, and the other under construction, CNA-II. At Río Tercero Dam Lake, CNE is presently in operation.

This report refers only to those studies that were carried out in order to qualify the above mentioned sites from the safety point of view. Other studies, covering different aspects of technical, economic, social and cultural nature associated with the nuclear power plants, have not been included in this report, although, obviously, they were considered at the time decisions related with the plants were made.

The investigations on natural external events considered floods, earthquakes and associated phenomena, such as geological faulting in the site vicinity and soil liquefaction, and extreme meteorological phenomena (as tornadoes). With regard to man induced external events, the potential for aircraft impact and explosions were analysed. In addition, detailed geotechnical investigations were performed to determine the soil parameters for stability verification and foundation design.

For the above mentioned sites, the most significant external events affecting the design basis were: seismic events for the Embalse site (earthquakes and geological faulting) and hydrological events for the Atucha site (extreme values of the Paraná River flooding potential).

The studies included the determination of the site characteristics that have influence on the effects of the plant operation on the environment, such as: meteorological and hydrological/hydrogeological characteristics affecting the dispersion in the atmospheric and hydrological aquatic media, respectively, the population distribution and the regional uses of soil and water.

For instance, the results of siting studies of the nuclear power plants were used in determining parameters required for the application of models describing radionuclide dispersion to the environment. Such models enabled the evaluation of dose exposure due to radioactive effluents released during normal operation.

Moreover, the information supplied by siting studies enabled to foresee the implementation of actions required to protect population from accidental situations. These steps were taken into account in the elaboration of the corresponding Emergency Plans.

17.3 NORMATIVE ASPECTS

A nuclear power plant construction may not be initiated without a previous construction license issued by the Regulatory Body, at request of the Responsible Organisation.

The Regulatory Body issues such license once the Responsible Organisation has demonstrated that the design of the nuclear power plant to be built complies with standards and other specific regulatory requisites for the selected site, taking into account the nuclear power plant-site interaction.

In summary, standards and regulatory requisites are applied to the installation safety related aspects as a whole. Because of this reason, the Responsible Organisation shall demonstrate, by means of appropriate siting studies, that the specific characteristics of the selected site that may influence the plant safety, as well as the potential impact of plant operation on the public and environment, verify the acceptance criteria established in the regulatory standards and they have been duly taken into account in the plant design bases.

It is for this reason that the regulatory system does not foresee the licensing of a site for location of a nuclear power plant as a separate process, previous to starting the plant design and construction.

In line with this approach, at the time of applying for the construction license the Responsible Organisation shall submit to the Regulatory Body all the documentation required to evaluate the radiological and nuclear safety of the installation to be built, including the site characteristics in relation to:

- Natural and man-induced external events affecting the installation safety.
- Dispersion of radionuclides to the environment, both in normal and accidental conditions.

Finally, the information that shall be submitted regarding this subject is established in a guide document named “Model for performing nuclear installation siting studies in Argentina”, which contains the criteria and theoretical-practical methods for the selection of suitable sites for nuclear power plants in a certain region.

17.4 STAGES OF SITE STUDIES

The site studies performed for both plants (i.e. CNE and CNA-I) were conducted in each case through three stages as follows: (1) survey of the region of interest, (2) selection of the candidate site, and, finally, (3) evaluation of the selected site.

For each plant, the first stage was the survey of a wide region with the purpose of screening (accepting or rejecting) those zones that could be candidates for location of a nuclear power plant. At this stage not only were safety considerations taken into account but also their economic and social aspects, as well as their evolution perspective during the plant lifetime were considered.

The site selection stages for CNE and CNA-I and CNA-II units comprised studies of several selected sites in order to demonstrate their acceptability, mainly from the safety point of view, even though the other aspects, as mentioned above, were also considered. At this stage, site related parameters required for the plant design against the effects of significant external events were preliminarily defined, as well as information that enable the evaluation of the effects that the plant operation produces on the public and environment was collected and analysed. As end product of this stage, the Embalse and Atucha sites were selected.

Finally, the evaluation stages comprised detailed studies aimed at determining the site related parameters for the design with respect to significant external events and for the assessment of the influence of the plants operation on the public and the environment. These studies were done in accordance with the knowledge and practice available at that time and were conducted during the construction and first operational stages.

17.5 SITING STUDIES PERFORMED

17.5.1 SITE STUDIES FOR ATUCHA I NUCLEAR POWER PLANT

Since the starting of Atucha I Nuclear Power Plant project, in the 60's, a number of studies and investigations were carried out in relation with the site characteristics, which were performed before and during the plant design and construction stage.

To this respect, the site survey stage – as mentioned in Section 17.4 above – was developed in the so called "*Gran Buenos Aires - Litoral*" region, which comprises parts of Buenos Aires, Santa Fé and Entre Ríos provinces. During this phase, two candidate sites were identified in the region: one, the Magdalena site located at the bank of the Río de La Plata River and to the south of Buenos Aires City, the other, the Atucha site, located at the right bank of the Paraná de las Palmas river and to the Northeast of Buenos Aires City. The latter was finally selected for locating the first nuclear power plant in the country.

The siting investigations are included in the studies entitled as "Pre-investment studies for the construction of a nuclear power plant in the Great Buenos Aires zone", which comprise technical, legal and economic aspects of the nuclear power plant to be built.

The CNA-I Safety Report was issued in 1972. It includes the analyses of problems related to the nuclear power plant site, with particular emphasis on aspects such as topography and human use of the environment, communication means, population distribution, and meteorological, geological and hydrological conditions, as well as the elimination of low activity wastes associated with the plant operation.

With the purpose of evaluating the impact on the environment due to CNA-I operation, several studies were carried out. Some of them include data that were collected before starting the nuclear power plant construction, while others were performed during that period. This enabled an evaluation of results as well as of the evolution of the impact on the environment due to the installation operation.

Moreover, studies about the climatological characteristics were carried out, such as: wind direction and frequency, atmospheric and river water temperature, relative humidity, precipitation values and atmospheric stability categories. A meteorological station (with a 100 m tower) was installed at the site around 1973, which is still in operation and it has been upgraded two times. Other studies were also performed, such as those to determine the population distribution at different distances from the site, as well as the land and water uses from these groups.

Finally, studies to determine dilution factors associated to the site were also carried out, as well as radioecological evaluations in vegetable species, in wild animals, in cattle breeding, agricultural local production and on sediments.

The report entitled "Nuclide transfer from water phase to sediments during the normal and extraordinary hydrological cycles of the Paraná de Las Palmas River", in relation to the radionuclide dispersion in water and their deposit on sediments should also be mentioned. This study was carried out in 1985, partially sponsored by IAEA.

17.5.2 SITE STUDIES FOR EMBALSE NUCLEAR POWER PLANT

The selection of a site for location of the second nuclear power plant in the country was carried out before year 1974 in the surroundings of Los Molinos Dam, Province of Córdoba. Because of the decision to build a plant with higher power and the non availability of sufficient cooling water for the heat sink, this zone was rejected.

In accordance with the new power requirements, the zone of Embalse de Rio Tercero was evaluated for location of the plant. Since 1974, the environmental characteristics of region were evaluated with the purpose of determining the hydrological aspects of Río Tercero basin and particularly those of the dam lake.

As result of this process, the site was selected at the coast of the lake and investigations and studies were carried out to evaluate the specific site related characteristics.

In particular, geotechnical, geophysical, geological and seismological studies, as well as meteorological investigations, were performed in order to determine the design basis parameters for natural external events. Information and data required for assessing surface faulting and seismic hazards were collected.

In this regard, the reports entitled as "Studies for the seismic evaluation of Córdoba Nuclear Power Plant siting", "Study of the Siting Seismic Risk" and "Final Report on the Re-evaluation of the Seismic Behaviour of Embalse Nuclear Power Plant in Córdoba" should be mentioned. All these studies were conducted during the seventies.

Hydrological characteristics and population water uses in direct consumption or irrigation were determined. As regards this item, not only the hydrological resources use at the time was evaluated but also projections contemplating the growing consumption trends due to expected economic development of the area were carried out.

The analysis of site meteorological conditions was carried out using data from the period 1972-1981, collected at the meteorological station located at the site, and operating since 1971.

A climatological study was performed and the following items were evaluated: environment temperature behaviour (daily and annual average), lake water temperature variation with depth, environment humidity, wind direction and speed, persistence of extreme temperature and humidity values, annual rain behaviour and its relation with wind direction.

Among the hydrological and meteorological analyses mention should be made of those performed in 1980 and 1983 respectively, named "Determination of the basic design tornado for Embalse Nuclear Power Plant, Córdoba" and "Environmental Study of Embalse Nuclear Power Plant Siting, Córdoba".

A man induced external events assessment was performed in 1982 in accordance with the Guidelines 50-SG-S5 and 50-SG-D5. This assessment covered aircraft crash hazards, external fires and explosions from solid substances. Although no designs basis for man induced events were derived from these studies, recommendations were given regarding air traffic, measures against forest fires and relocation of the public road in front of the plant.

Finally, it should be mentioned that studies referred to population distribution, housing, cattle breeding, fishing and agricultural activities and other characteristics of the local ecosystem were also carried out, enabling the evaluation of the effects of the plant operation on it; besides, it is important to consider the information on CNE site available in "CNE Safety Report" versions of 1985, 1986 and 1993.

17.5.3 **SITE STUDIES FOR ATUCHA II NUCLEAR POWER PLANT**

Due to the fact that CNA-II unit is located in the same site as CNA-I's, a lot of specific information was available at the time of CNA-II design stage. This was the result of continuous studies that are being carried out for CNA-I since it first began operating, particularly about hydrological, extreme meteorological phenomena and atmospheric dispersion, and population distribution aspects, as well as to the nuclide transfer mechanisms models, both through atmospheric and aquatic media.

Additionally, specific site related studies were carried out to determine the design basis parameters for the new unit.

Extensive geotechnical investigations were performed to define the soil characteristics for stability verification and foundation design, as well as geophysical investigations and studies for determining the seismic hazard in accordance with new criteria and data. Thus, the report entitled as "Seismic Study of CNA-II Nuclear Power Plant Siting" reflects the results of such investigations, which were carried out in 1981 by the "Argentine Nuclear Company of Electrical Power Plants". It should also be mentioned that the corresponding chapter of the "Preliminary Safety Analysis Report" was issued in 1981 and include all information about the site.

Other natural external event that has been included in the design basis of the plant is tornado. Thus, the tornado generated projectiles and pressure loads were taken into account in the design of the plant. It should also be mentioned the operation of the meteorological station (with a 100 m tower) at the site which provides specific site micro-meteorological data.

Man induced external events are also considered in the plant design basis. In this regard, an explosion pressure wave corresponding to deflagration of a gas cloud is defined, which is duly taken into account through appropriate layout and structural aspects.

Other site related studies to be mentioned are the specific population census in the plant surroundings (completed by regional and national census) or those related to the cattle breeding and agricultural production around the plant.

Moreover, values of tritium concentration measurements in environmental compartments were confronted against the corresponding theoretical values, enabling the dilution factors validation.

Finally, it should be mentioned that monitoring on vegetables, atmospheric and hydrological sediments, cow milk from the region, fish, water, etc., are carried out, which enable the verification of transfer parameters that are used in the evaluation of individual and collective dose associated with CNA-II operation.

17.6 SITE RE-EVALUATION

The site characteristics that may potentially affect safety may require to be re-evaluated during the plant lifetime, because new data become available or due to changes in the criteria and methods in accordance with advances in the recognised practice and the evolution of the state of the art in the subject. As end product of this re-evaluation, new parameters may be defined for the external events and, consequently, the plant safety needs to be verified for the new conditions.

In the country, re-evaluations concerning the most significant external events were carried out for both Atucha I and Embalse nuclear power plants. In particular, the hydrological characteristics for the Atucha site and the seismic hazard for the Embalse site were re-evaluated after the construction of the plants.

In the case of the Atucha site, the impact of potential failure of the dam located upstream in the Paraná River was evaluated and, as a consequence, the extreme value of the flooding level was modified and several changes were introduced in the cooling water inlet structures for CNA-II. The seismic hazard was also reviewed for this site and the original design of this unit (whose construction is almost completed) is under verification in order to cope with the new seismic loads.

In the case of the Embalse site, the re-evaluation of the seismic hazard resulted in higher seismic loads for the severe earthquake level and, consequently, as result of the structural response analysis, a number of modifications were carried out to safety related components and systems, before plant commissioning once it was built (see Section 14.1.2.2). Presently, a programme for updating and upgrading plant seismic instrumentation, including the preparation of a plant operating procedure for responding to the occurrence of a seismic event, is being implemented.

Besides, the following studies were carried out:

- Analysis of the effect of earthquakes having a maximum horizontal aceleration value between 0.17 and 0.34 *g* in the Embalse Dam (1987).
- Analysis of the hydrological consequences of the functioning of the Rio Grande hydro-electrical complex on CNE's operation. This complex is located by Rio Grande River, tributary of Embalse Dam, some 20 km up river from the nuclear power plant (1985).

At present, the Regulatory Body is developing several projects related to site re-evaluation. The purpose of one of them is to develop a new calculation code for the evaluation of the consequences of accidental nuclide discharges to the atmosphere, in the case of CNE, which will be more adequate for the geographic characteristics of the zone than the one used now. The purpose of another one is to analyse from a probabilistic point of view, the consequences of radionuclide emission to the environment due to eventual accidents in CNA-I and CNE, using the COSYMA code.

Other project under implementation is the re-evaluation of extreme meteorological phenomena (mainly tornadoes) for both sites with the purpose of verifying the original design criteria in accordance with the new data available and updated hazard modelling and evaluation techniques.

17.7 COMPLIANCE WITH THE OBLIGATIONS IMPOSED BY THE CONVENTION

From the preceding considerations, it becomes evident that in the country the significant factors related to nuclear power plant site that could affect its safety during their lifetime have been evaluated. Moreover the radiological impact on the public and environment due to their operation has also been evaluated. At the same time, the main site-related factors that ensure the continued safety acceptability of the nuclear power plants have been re-evaluated.

Therefore, the country complies with the obligations imposed by the Article 17 of the Convention on Nuclear Safety.

ARTICLE 18

DESIGN AND CONSTRUCTION

18.1 INTRODUCTION

The Regulatory Body has issued standards which cover the necessary design and construction aspects in order to prevent accidents as well as to mitigate their radiological consequences if they occurred.

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

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

Besides, requirements taking into account the prevention of eventual component degradation, maintenance of safety systems reliability levels, and implementation of an emergency plan are included in the respective operating licenses.

18.2 DESIGN AND CONSTRUCTION

CNE nuclear power plant was designed and built in such way as to count with levels and reliable protection methods against accidental release of radioactive materials (defence in depth principle), with the purpose of preventing accidents and mitigating their radiological consequences in case of occurrence. CNA-I nuclear power plant, designed before the defence in depth principle was first stated, also complies with the basic criteria associated with the principle.

Such basic safety principles are included in the applicable regulatory standards, which are detailed in Tables 18.1, 18.2 and 18.3. The purposes of each of such standards are the following:

AR 3.1.3 standard establishes the general conditions that power reactors shall fulfil in order to prevent the occurrence of accidents as well as mitigate their radiological consequences if they occurred.

AR 3.2.1 standard sets the general criteria safety systems shall comply with.

AR 3.2.3 standard contains those criteria related to fire protection or events generated by fire and explosions that could concern radiological or nuclear safety.

Table 18.1

Standard	Name
AR 3.1.3	Radiological Criteria Related to Accidents In Nuclear Power Plants.
AR 3.2.1	General Safety Criteria in the Design.
AR 3.2.3	Fire Protection.
AR 3.3.2	Heat Removal Systems.
AR 3.3.3	Primary Pressure Circuit.
AR 3.3.4	Fuel Element Behaviour in the Reactor.
AR 3.4.1	Protection System and Safety Related Instrumentation.
AR 3.4.2	Shutdown Systems.
AR 3.4.3	Confinement Systems.
AR 3.5.1	Emergency Electric Power Supply.
AR 3.6.1	Quality System.
AR 3.10.1	Protection against Earthquakes.

AR 3.3.2 standard determines the requisites for an adequate fuel element cooling in the core.

AR 3.3.3 standard contains the necessary requirements to preserve the primary pressure circuit integrity under any operational and failure condition.

AR 3.3.4 standard requires the fuel elements to be designed and manufactured minimising probability and amount of radioactive material release.

AR 3.4.1 standard sets, among other items, the following:

- The protection system shall be automatically activated.
- The operator's action should not be necessary during a suitable time interval after such activation.
- The operator may initiate protection procedures but he shall not avoid the necessary operation of the protection system.
- The protection system shall comply with the single failure criterion.

AR 3.4.2 standard specifies design characteristics of reactor shutdown systems.

AR 3.4.3 standard sets design characteristics of confinement systems, particularly regarding barrier number, radioactive material retention capability, accidental and normal loads, confined atmosphere leakage rate and the corresponding verification tests.

AR 3.5.1 standard contains guidelines to ensure, by means of an adequate design, an effective and reliable emergency[1] electric power supply required to preserve safety during operational and accidental situations.

[1] In the country this system is called "*Alimentación eléctrica esencial*" (Essential Electric Power Supply). See AR 3.5.1 Standard. ANNEX 1.

AR 3.6.1 standard sets the minimum requisites in order to develop, establish and implement a quality system capable of including all the aspects related to radiological and nuclear safety.

AR 3.10.1 standard establishes the design-related criteria applicable to protection against earthquakes.

Regulatory standards establish that those technologies incorporated to the nuclear power plant design and construction, shall be validated by experience or verified through tests and analyses. The standards shown in Table 18.2 are applicable.

Table 18.2

Standard	Name
AR 3.2.1	General Safety Criteria in the Design.
AR 3.3.1	Reactor Core.
AR 3.4.1	Protection System and Safety Related Instrumentation.
AR 3.4.3	Confinement Systems.
AR 3.5.1	Emergency Electric Power Supply.

In what follows the purposes of each of such standards are mentioned:

AR 3.2.1 standard specifies the conditions to be fulfilled by structures, systems and components design methods regarding the tolerance in data uncertainty, tests they shall be submitted to, reliability and redundancy.

AR 3.3.1 standard sets the requirements to be satisfied by design data, particularly those related to the reactor physics and thermohydraulic behaviour.

AR 3.4.1 standard sets that protection system components shall have proven reliability and effectiveness, and determines the minimum redundancy of such systems.

AR 3.4.3 standard determines redundancy, diversity and segregation features that shall have confinement systems to ensure the necessary reliability to comply with AR 3.1.3 standard.

A.R. 3.5.1 standard establishes that reliability and availability required for the emergency electric power supply system shall be specified, and that the means used to fulfil such requirement shall be described. Besides, it establishes that it must be demonstrated by means of calculations, and verified with the corresponding tests and measurements, that the emergency electric power supply system will be available for every demand condition (including the effects of different load states). In Table 18.3 a list of the regulatory standards applicable to nuclear power plants design, establishing requirements for a reliable, stable and easy control operation of these installations, with special consideration of human factors and the man-machine interface, is shown:

Table 18.3

Standard	Name
AR 3.2.1	General Safety Criteria in the Design.
AR 3.4.1	Protection System and Safety Related Instrumentation.

In what follows the purposes of each of such standards are mentioned:

AR 3.2.1 standard defines significant aspects related to the operator's role. Among them it should be mentioned that the operator shall know any information required in order to take safety related decisions, and that it shall be forbidden for him to act during the time interval immediately subsequent to the occurrence of an accident initiating event, after this period, the operator shall be able to take any measure in favour of safety, but he will not be able to avoid the action of any protection system.

AR 3.4.1 standard sets the characteristics man-machine interaction shall have regarding the reactor protection and instrumentation design.

18.3 NUCLEAR POWER PLANTS COMPLIANCE WITH REGULATORY STANDARDS

Some regulatory standards were issued after the construction of the nuclear power plants CNA-I and CNE, so that the regulatory body did not ask for their immediate application. Nevertheless, these standards are already been fulfilled or they are being implemented.

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

The primary circuit integrity for both normal and accidental conditions, is preserved considering the effect of anchorages, connections, internal and external loads and deformations caused by thermal, mechanical and irradiation effects.

The CNA-I and CNE nuclear power plants have a containment representing the last confinement barrier, designed to support loads due to different accidental situations both from internal and external initiating events.

The containment system design criteria are specially referred to the number of barriers, retention capability of radioactive material, behaviour under normal and accidental loads, leakage rate to the atmosphere and the corresponding verification tests.

The shutdown systems (control rod insertion and liquid poison injection) insure the reactor shutdown in normal and accidental situations, keeping such safe state during the necessary time period.

The emergency electric power supply design criteria allow the preservation of an adequate safety level under normal and accidental conditions. It also includes the independence, redundancy, physical separation and diversity criteria. External events such as fire and missiles are also considered.

18.3.1 **ATUCHA I NUCLEAR POWER PLANT**

Reactor safety systems design and confinement barriers preventing fission products release, such as fuel elements claddings, primary circuit and reactor containment, comply with the criteria established in AR 3.2.1 standard. Moreover, the safety systems design complies with the single failure criterion as well as with segregation and diversity. This latter also applies to all those systems which may require it.

The core heat removal system design complies with the requirements of AR 3.3.2 and AR 3.3.3 standards under normal operation (heat transport primary system and shutdown cooling system) and during hypothetical accidental situations (emergency core cooling system).

Both CNA-I shutdown systems design comply, in general, with the criteria established in AR 3.4.2 standard, particularly in what concerns diversity, redundancy and reliability.

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

- *The containment system:* this system is constituted by a steel sphere of approximately 50 m in diameter wrapped up by a second safety cover of concrete, as its external shield. The system includes several penetrations, air locks and the isolation contention sub-system.
- *Radioactive material removal system in case of accident:* this system is located between the steel sphere and the external shield and operates by passing air through carbon and absolute filters.

In case either a LOCA or a breakage in the secondary system within the containment takes place, it is foreseen that its resulting pressure shall not exceed the design values (2.8 kg/cm^2). On the other hand, the leakage rate design value of the atmosphere confined by the contention is 0.5% per day of free volume.

The design pressure and the containment leakage rate were verified during tests, according to what is established in Section 3.8 of the operating license. The tested pressure was 3.1 kg/cm^2, while the measured leakage rate was much smaller than the corresponding design value.

CNA-I design complies, in general, with the requirements of AR 3.2.1, AR 3.3.1, AR 3.4.1 and AR 3.4.3 standards, particularly regarding the uncertainty data boundary, and the application of safety concepts valid when its design was developed, such as redundancy, diversity, etc.

Such verification was obviously indirect, since the purchase contract specified that the components, designed and manufactured in the Federal Republic of Germany, had to comply with the requirements of German standards and that the nuclear power plant had to be licensable in that country.

On the other hand, methods and calculation tools compatible with the state of the art in those times and verified through operation experience were used in the core design.

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

Taking into account the state of the art regarding information processing and report systems at the time the nuclear power plant was designed, AR 3.4.1 standard requirements concerning man-machine interface are in general fulfilled. Particularly, during an appropriate time interval after the automatic activation of a safety system, no action is required from the operator who, on the other hand, is not able to avoid or interrupt its operation. Nevertheless, the operator may initiate other safety actions.

As already mentioned in Section 11.7.1, a graphic interactive simulator, used for training personnel to be incorporated in the installation, was implemented.

18.3.2 **EMBALSE NUCLEAR POWER PLANT**

The reactor safety system design and the contention barriers preventing fission product release, such as the fuel pellet itself, the fuel element clad, the heat transfer pressurised circuit, and the reactor building comply with the requirements of AR 3.3.2, AR 3.3.3 and A.R. 3.3.4 standards.

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

The confinement barrier required by AR 3.4.3 standard in CNE is constituted by the following systems:

- *Containment system:* this system is constituted by the building reactor structure, its penetrations, airlocks and isolation contention devices.
- *Pressure suppression system:* this system is constituted by the dousing system and the building air coolers.
- *Fission product removal system:* this system is constituted by the ventilation and the reactor building atmosphere steam recovering system.

In case either a LOCA or some secondary system component breakage inside the reactor building took place, the dousing system keeps its pressure below the design values (1.25 kg/ cm^2). According to what is established in point 3.9 of the operating license and in AR 3.2.1 standard, periodic containment leak proof tests shall be at least carried out each five years.

CNE design complies with AR 3.2.1, AR 3.3.1 and AR 3.4.3 standard requirements, particularly regarding the uncertainty data boundary, and the application of safety concepts valid when it was designed, such as redundancy, diversity, etc.

On the other hand, methods and calculation tools compatible with the state of the art in those times and verified through operation experience were used in the core design.

The structural integrity of the confinement system and the leakage rate of the confined atmosphere were verified by experimental tests with satisfactory results (these tests are part of the periodic test program of the installation). The value of the differential pressure during last test (December 1995) was acceptable and the leakage rate value was lower than its corresponding design value, i.e. 0.5% per day of free volume.

AR 3.2.1 standard criteria related to the operator performance, are in general, fulfilled. Concerning the intervention in case of accidents, the operator must always take provisions to avoid a situation that could affect the nuclear power plant safety, but he should not avoid the necessary operation of safety systems. The operation of the low pressure emergency cooling system can be interrupted, since it is designed to operate during long time intervals, and under certain circumstances it may be necessary to discontinue its operation.

Taking into account the state of the art regarding the information processing and report systems at the time the nuclear power plant was designed, AR 3.4.1 standard requirements related to man-machine interface are in general fulfilled. In addition and according to Section 11.7.2 contents, in 1994 a simulator, used for training of personnel to be incorporated in the installation, was implemented.

18.3.3 **ATUCHA II NUCLEAR POWER PLANT**

The conceptual engineering and basic design of the reactor safety systems and the barriers to accidental release of fission products to the environment, such as fuel element cladding, pressurised circuit of heat transport and reactor containment comply with the criteria established in AR 3.2.1 standard. Moreover, the safety system design complies both with the single failure criterion and the segregation and diversity principles, as they use a two-out-of-four logic. This last one is also applied to every system requiring it.

The design of the core heat removal system during the future normal operation (heat transport primary system and shutdown cooling) and in case of hypothetical accidental events (emergency core cooling system), comply with AR 3.3.2 and AR 3.3.3 standards. A report of a probabilistic safety evaluation shows that AR 3.1.3 standard is fulfilled.

The design of the two CNA-II shutdown systems verify, in general, the criteria established in AR 3.4.2 standard, particularly in what concerns to diversity, redundancy and reliability concepts.

CNA-II has the following systems constituting the confinement barrier required in AR 3.4.3 standard:

- *Containment system*: it is constituted by a steel sphere covered by a second safety shell made of concrete, having the additional exterior shielding function. The system has several penetrations, airlocks and containment isolation devices.
- *Radioactive material removal system*: it is located between the steel cover and the external shielding and is acted by the air passing through carbon and absolute filters.

It should be emphasised that CNA-II's design has been conceived taking into account all the improvements emerged from the operational experience in CNA-I, such as the reduction of the Stellite-6 covering surface in moderator-primary circuit components and the design modifications cooling channels.

18.4 COMPLIANCE WITH THE OBLIGATIONS IMPOSED BY THE CONVENTION

In the country, nuclear power plants have been designed and constructed in such a way as to have several reliable protection levels, in order to prevent the release of radioactive materials to the environment, prevent accidents and mitigate their consequences in case they occurred.

Therefore, the country complies with the obligations imposed in Chapter 18 of the Convention on Nuclear Safety.

ARTICLE 19
OPERATION

19.1 INTRODUCTION

The Regulatory Body authorised nuclear power plants commercial operation on the basis of the judgements mainly supported by both safety assessments and commissioning follow-up results at the installations.

Once in operation, nuclear power plants are operated by the Responsible Organisation according to what is established in the operating license and operational limits and conditions set in the Safety Report and the Policies and Principles Manual. This last document is based on the technical design specifications of the installation and the operating experience.

Regulatory Body inspectors verify that the three above mentioned documents are fulfilled. Moreover, as part of routine inspections, resident inspectors audit and control procedures fulfilment and regular test performance, in-service inspections, programmed maintenance and any other safety related activity.

The Primary Responsible is supported by an engineering section giving part of the technical support needed for the nuclear power plant operation. Besides, the Responsible Organisation also has an engineering division satisfying some of the technical support installation needs, and in order to cover other required services, domestic or international contractors are used.

The process, through which the installation operational experience feedback is carried out, both at the Primary Responsible and the Responsible Organisation level, must comply with the operating license requirements and AR 3.9.2 standard as well as with every other regulatory requirement.

In the same way, the Regulatory Body participates in the IAEA Incident Reporting System. Furthermore, it is part of the IAEA International Nuclear Event Scale. With the purpose of taking advantage of operational experience, since 1994 the Regulatory Body is also part of the co-operation group among regulatory authorities of countries having CANDU type nuclear power plants. At the same time, the Responsible Organisation of nuclear power plants are part of the incident information systems established by the operators organisations: Candu Owners Group (for CNE) and World Association of Nuclear Operators (for CNE and CNA-I).

The feedback process of operational experience of domestic nuclear power plants involves the following entities: Responsible Organisation, Regulatory Body, designers, component suppliers and international organisations dedicated to information distribution.

The Responsible Organisation must inform the Regulatory Body about significant events produced in the installations within established time limits. Specialised professionals of the Regulatory Body analyse the available information, and according to the conclusions reached, they issue requirements to the Responsible Organisation. The description of the event occurred, the Regulatory Body requirement and the corrective actions taken by the Responsible Organisation are communicated to the international organisations, which are consequently responsible of distributing them to the international community.

Finally, it is mentioned that nuclear power plants have programs for the management of radioactive wastes generated during their operation. The programs include low and medium radioactive waste treatment and its subsequent storage.

19.2 INITIAL AUTHORISATION

In Section 14.1.1.3 those aspects related to CNA-I commissioning are developed.

CNE initial authorisation was issued according to the requirements established in AR 3.8.1 and AR 3.8.2 standards. The first one is related to pre-nuclear commissioning and establishes that the Responsible Organisation must have a commissioning program and an organisation to carry it out. The pre-nuclear commissioning program numbers those tests required in order to demonstrate the safe operation of the nuclear power plant.

AR 3.8.2 standard also establishes that the Responsible Organisation must have a nuclear commissioning program and an organisation to carry it out. The standard also establishes that the Responsible Organisation must appoint an ad-hoc committee for the nuclear commissioning follow-up, constituted by qualified personnel having experience in design, construction and operation of nuclear power plants. The ad-hoc committee has the main responsibility of evaluating each of the stages the commissioning program is divided into, and authorises the transition from one stage to the other.

During pre-nuclear and nuclear commissioning stages, the Regulatory Body verified that the Responsible Organisation complied with the mentioned standards (see Sections 14.1.2.2 and 14.1.2.3).

19.3 ATUCHA I NUCLEAR POWER PLANT

19.3.1 CONDITIONS FOR OPERATION

The conditions for the authorisation of the commercial operation of CNA-I were established in the operating license. The main requirements for the nuclear power plant, such as maximum reactor thermal power, authorised discharge limits, communications to the Regulatory Body of the occurred significant events, etc. are explicitly contained in the license, or referred to other mandatory documents.

Initially there was at CNA-I no specific document referred to operational limits and conditions, as there are in most of the nuclear power plants. The existing information -at that time spread out in different documents, such as the Safety Report, the Operating Manual, and the Maintenance Manual- has been collected in the Policies and Principles Manual. Besides, one of the requirements of the last operation license issued in 1990 (see Section 6.2.1.1) concerns the inclusion of a special chapter dedicated to operational limits and conditions in the Operating Manual.

19.3.2 OPERATIONAL LIMITS AND CONDITIONS – MAINTENANCE, INSPECTIONS AND TESTS

The Policies and Principles Manual of CNA-I establishes the ranges of valid values some plant operational parameters shall be within, other specifications as well as the organisation requirements that shall be satisfied in order to ensure a safe operation.

The operational parameters concern mainly reactor power, core reactivity control, heat transport systems, refuelling and secondary system related parameters. The specifications referred to the operating organisation comprise, among others safety related subjects, personnel licensing, minimum staff in plant and control room (see AR 3.9.1 Standard), the Safety Advisory Internal Committee activities and the communication of significant events to the Regulatory Body.

CNA-I has preventive maintenance and in-service inspection programs, which include scope, planning, implementation and control of the preventive, predictive and corrective maintenance activities (see Section 14.3). All these activities are performed according to a set of procedures and manuals, which are part of the mandatory documentation required in the operating license.

The surveillance program including in-service inspection activities related to significant components, equipment and systems are routinely carried out, mainly involving the following: pressure vessel; primary, moderator and volume regulation systems, as well as steam generators and moderator heat exchanger tubes.

19.3.3 **OPERATIONAL PROCEDURES IN NORMAL AND ACCIDENTAL CONDITIONS**

Most of the operational procedures, either in normal or accidental conditions, are included in the Operating Manual. Such document has three parts:

- *The first part* has general plant descriptions, design parameters and operation mode.
- *The second part* has specific operation information; basically instructions of the **G** type, to modify the installation operational state, and instructions of the **J** type, to perform infrequent hand made actions.
- *The third part* includes the manual of warnings and alarms of all the installation boards, instructions of the **E** type for emergency cases and instructions of the **T** type for abnormal cases.

19.3.4 **TECHNICAL SUPPORT**

CNA-I operating organisation has its own engineering section. Such section is complemented with the Responsible Organisation technical services, which include specific subjects such as instrumentation-control and civil engineering, having a qualified staff of operators who normally give support before and during the scheduled outages.

In some issues like non-destructive tests, materials, corrosion and water chemistry treatment, the nuclear power plant operating organisation asks the National Atomic Energy Commission for service and specialised advice (technical support). Frequently it has also used the services of INVAP S.E. (an Argentine technology organisation dedicated to high technology projects). It has also used and will keep on using, if needed, the advice of Siemens - Kraftwerk Union AG, responsible for the nuclear power plant design and construction.

19.3.5 **CRITERIA AND REGULATORY REQUIREMENTS FOR THE ACCIDENT INFORMATION SYSTEM**

One of the main concerns of the Regulatory Body is the significant event occurrence and the actions related with it, considered part of the profit from operational experience in nuclear power plants.

To this respect, AR 3.9.2 standard sets the basic criteria concerning definitions, event communication modes to the Regulatory Body, and event analysis. On the other hand, the operation license sets particular conditions referred to the subject and some specific requirements have been issued concerning it.

19.3.6 ON-SITE IRRADIATED FUEL AND RADIOACTIVE WASTE MANAGEMENT

CNA-I irradiated fuel elements are temporarily stored in the installation decay pools. Such pools, located in the Fuel Storage Building, have enough capacity to store the irradiated fuel produced by the nuclear power plant during its full power operation life and a complete core discharge.

During CNA-I operation different radioactive wastes are generated with characteristics, origin and management described in what follows.

Ionic exchange resins, used to purify the primary coolant water, are medium activity radioactive wastes and are stored in decay tanks located in the Fuel Storage Building. The volume generated is about 0.8 m^3/year.

Radioactive sediments generated in the primary circuit, are cemented and then sent to National Atomic Energy Commission low activity waste management plant, located at the *Centro Atómico Ezeiza* (Ezeiza Atomic Centre).

In the installation there are several systems carrying out functions such as volume regulation, ventilation and coolant cleaning using different types of filters. Once the filters have been replaced, they are managed and stored in deposits located in the nuclear power plant, destined to such purpose.

Other compressible and non-compressible radioactive wastes are also generated. The compressible wastes such as gloves, papers, and contaminated cloths, are reduced in volume in a ratio of 5:1 and are located into 200 litres specially designed drums for their disposal in surface trenches destined to low activity wastes. The non-compressible wastes such as tools and pieces of contaminated metal or wood coming from maintenance or cleaning processes are cemented into similar drums. The drums are temporarily stored in installations conditioned to such purpose in the nuclear power plant, and they are then transferred to National Atomic Energy Commission radioactive waste management plant located at the Ezeiza Atomic Centre to complete their final disposal.

19.4 EMBALSE NUCLEAR POWER PLANT

19.4.1 CONDITIONS FOR OPERATION

The conditions for the initial authorisation of commercial operation of Embalse nuclear power plant have been mainly established in the operation license. In fact, in the license, the essential requirements for the installation operation such as maximum reactor thermal power, limits of authorised discharges, communication of the occurred significant events to the Regulatory Body, etc. are explicitly contained or refer to other related documents.

Another conditioning requirement for CNE commercial operation was CALIN 122/84 document (see Section 6.2.2.5).

19.4.2 **OPERATIONAL LIMITS AND CONDITIONS – MAINTENANCE, INSPECTIONS AND TESTS**

Since its commissioning, CNE has a Policies and Principles Manual where limits and conditions for the installation safe operation are established. Such limits and conditions mainly arise from the Canadian experience on CANDU type reactor operation, transferred to CNE. The Policies and Principles Manual is also the reference framework for most of the nuclear power plant operational procedures.

CNE has preventive maintenance and in-service inspection programs, which include scope, planning, implementation and control of the preventive, predictive and corrective maintenance activities. All these activities are performed according to a set of procedures and manuals, which are part of the mandatory documentation required in the operating license (see Section 14.3).

The surveillance program including in-service inspection activities related to significant components, equipment and systems are routinely carried out, mainly involving the following: pressure tubes; primary, moderator and volume regulation systems, as well as steam generators.

19.4.3 **OPERATIONAL PROCEDURES IN NORMAL AND ACCIDENTAL CONDITIONS**

Most of the normal activities carried out at CNE are considered in procedures applied either in normal operation or accidental situations (see Sections 19.5, 19.6 and 19.7).

19.4.4 **TECHNICAL SUPPORT**

CNE operating organisation has its own engineering section complemented by the Responsible Organisation technical service groups. These include specific subjects such as instrumentation-control and civil engineering and have a qualified staff of operators who give support before and during the scheduled outages.

Besides, National Atomic Energy Commission service and specific advice are used by the nuclear power plant operating organisation to cope with issues like non destructive tests, materials, corrosion and water chemistry. It has also frequently used the services of INVAP S.E. (Argentine technology organisation dedicated to high technology projects). Additionally, it has used and will keep on using according to its needs, the advice of Atomic Energy of Canada Limited, responsible for the design and construction of the nuclear power plant, as well as the companies that operate CANDU type reactors, with which there is an active experience exchange.

19.4.5 CRITERIA AND REGULATORY REQUIREMENTS FOR THE ACCIDENT INFORMATION SYSTEM

The information already supplied for CNA-I also corresponds to this section (see Section 19.3.5).

19.4.6 ON SITE IRRADIATED FUEL AND RADIOACTIVE WASTE MANAGEMENT

Irradiated fuel elements are stored in pools that are part of the installations, having a capacity which is limited to ten years of full power operation and enough space so as to store a number of fuel elements corresponding to a complete core. After a decay period of 6 years, the fuel elements are transferred to special silos located within the nuclear power plant site for their transient dry storage (see Section 6.2.2.4).

The wastes generated in the water purification process of the primary circuit are basically exhausted ionic exchange resins. About 12 m^3/year of these resins are managed as medium activity wastes. They are stored in two decay deposit of 200 m^3 capacity each, also located within the nuclear power plant site.

During CNE normal operation, other radioactive wastes are generated. Solid low radioactive wastes, are in most cases compressible elements such as gloves, cloths, and contaminated papers. Such wastes are reduced in volume and then introduced in special 200 litres drums. The non-compressible wastes (contaminated elements such as pieces of metal and wood, tools, etc.) are cemented and also introduced in similar drums. The drums are temporarily stored in installations conditioned for such purpose within the nuclear power plant site, to be afterwards sent to National Atomic Energy Commission radioactive waste management plant located at the Ezeiza Atomic Centre to complete their management.

Solid medium radioactive wastes are mainly composed of mechanical filters coming mainly from the primary system of heat transport, moderator system, loading machine and pools. These filters, the cartridges of which are taken out, are deposited in bags or drums inside the deposits destined to that purpose. On the other hand, it is important to emphasise that the plant has five wells for the disposal of solid radioactive wastes.

19.5 OPERATIONAL EXPERIENCE FEEDBACK

Any event implying operation anomalies, be it insignificant or the most severe that may occur in any installation, has to be analysed aiming to obtain knowledge in order to minimise its probability of occurrence and diminish its consequences as much as possible.

Taking into account this principle regarding safety, the Regulatory Body has issued regulations the Responsible Organisation has to put into practice and comply with. In this sense the Regulatory Body has not only been limited to establish the specific standards and particular requirements according to the provisions in Section 19.3.5 and 19.4.5 but it is also in charge of the co-ordination in the country, of the IAEA Incident Reporting System.

19.5.1 OPERATIONAL EXPERIENCE

In order to improve operational safety in CNA-I and CNE nuclear power plants, a periodic analysis of their operational experience, and, to a smaller extent, an assessment of other nuclear power plants operational experience, are carried out.

The result of the identification of direct and root causes of the selected events is transformed into corrective actions implemented in the nuclear power plants, their effectiveness evaluated, transferred to the other plant, to the Regulatory Body and to the international nuclear community through the Incident Reporting System.

19.5.1.1 Feedback of the own operational experience

Both nuclear power plants have, as part of their internal organisation, an arrangement for the analysis of the operational experience, and carry out the resulting improvements and the information of results.

In both nuclear power plants the following internal events are detected, recorded and analysed:

a) Significant Events, defined according to the criteria set in standard AR 3.9.2

b) Unforeseen outages[1].

c) Minor events (CNA-I) or reportable events (CNE)[2].

Although this task has particular characteristics for each plant, the final result of the management of these events is similar. Each type of event is selected,

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[1] In CNE these events are included in the group of the so called "significant events". In CNA-I they are considered separately.

[2] The minor or reportable events are not directly related to safety, but they may eventually be precursors of significant events.

analysed and if corresponds, the corrective action identified and implemented, and the information distributed in the plant or in other plants according to specific procedures.

As regards significant events, the plant procedures comply with the corresponding Regulatory Body standards AR 3.9.1 and AR 3.9.2. These standards establish criteria for the selection, analysis and information of the significant events occurred in an installation.

In CNE, the reportable events are analysed according to a procedure, by the section in which it was originated. The recommendations or corrective actions, their implementation and distribution are a responsibility of the section involved.

In CNA-I, any person belonging to the installation can originate the notification of a minor event. A committee constituted by members of the different sections (Operation, Engineering, Mechanical Maintenance, etc.) evaluates these events, proposes corrective actions and follows their implementation and distribution.

All the operational incidents, significant and minor events, their corrective actions and their follow up are recorded.

Every event implying an unforeseen outage and/or a violation of limits and operational conditions established, must in addition be evaluated by the Safety Advisory Internal Committee of the plant, according to what is established in standard AR 3.9.1. Its conclusions and recommendations are written down in minutes signed by the participants.

In addition, the Technical Revision Committee, independent from the installation, must analyse the importance of the operational incidents foreseen, and the significant events occurred. Its conclusions and recommendations are recorded in minutes signed by the participants.

The significant events are communicated to the Regulatory Body according to standard AR 3.9.1 and afterwards an analytical report is issued according to dates and style established in the plant operating license.

The Regulatory Body informs the IAEA Incident Reporting System about the significant events occurred in the nuclear power plant, in order to enable the contribution of data about operational experience to other nuclear power plants, and soon notifies the international community of the occurrence of a significant event together with its category according to the IAEA International Nuclear Events Scale System.

Finally, and as an example of the operational experience feedback of the nuclear power plants in operation in the country, the transfer to CNA-II of those problems detected during CNA-I operation should be mentioned (see Section 6.2.3).

19.5.1.2 **Feedback of the operational experience of other nuclear power plants**

At the beginning of CNA-I operation, its designer, Siemens - Kraftwerk Union AG, played an important role in the transmission of operational experience of the German PWR, applicable to that nuclear power plant.

CNE has had, since the beginning of its operation, a fluent communication with other CANDU plants of similar design, such as Point Lepreau, Gentilly-II, Wolsung-II, in order to exchange operational experience. Moreover, it is member of the Candu Owners Group since its creation.

Presently, both CNA-I and CNE nuclear power plants receive information from the following databases:

- CANDU Owners Group.
- World Association of Nuclear Operators.
- IAEA International Reporting System.

The processing of information provided by the different sources is heterogeneous and not always profitable, as it is essentially dependent of the characteristics of the plant's design.

CNE uses Candu Owners Group databases as part of their usual working activities. The Engineering Section is in charge of carrying out a first selection of the information and transmitting it to the sections involved. Several corrective actions have been implemented as a consequence of the information received via Candu Owners Group. On the other hand, CNE provides Candu Owners Group a periodic report of its significant events.

CNA-I has been mainly using the World Association of Nuclear Operators database since 1996. The collection, selection and classification of information have been systematised.

In order to fulfil a requirement posed by the Regulatory Body, the Responsible Organisation is presently elaborating a "Program for the operational experience management", that will enable the improvement of the present system, exchanging experience of each plant on the subject, as well as taking better advantage of the international experience. The program shall include a methodology for the selection of applicable events for each plant, and will enable the systematisation and improvement of the analysis of root causes, the implementation and follow up of corrective actions, and an extension of the systematic analysis of all the available databases. In addition, the elaboration of an organisational structure will enable the optimisation of the event analysis and its feedback to the installation.

The pressure tube inspection program carried out at CNE nuclear power plant is an example of the profit of operational experience feedback of foreign installations (see Section 6.2.2.2).

19.5.2 LESSONS LEARNED FROM THREE MILE ISLAND AND CHERNOBYL ACCIDENTS

The accident of Three Mile Island occurred in March 1979. The radioprotection head of CNA-I Nuclear Power Plant, the only nuclear plant in operation in the country at that time, was sent to the site in order to get in touch with the circumstances related to the event, and the corrective actions taken afterwards.

On the other hand, Siemens - Kraftwerk Union AG made an exhaustive analysis of the accident, with the purpose of detecting those design aspects of the nuclear power plants built by them that could lead to operational situations similar to those occurred during the accident suffered by the American nuclear power plant. In particular, the specialists belonging to Siemens - Kraftwerk Union AG and the Responsible Organisation analysed, for the case of CNA-I, a series of modifications successively implemented at the installation. Such modifications were:

a) The reactor trip signal related to low water level in the steam generators was reset to a new level (from the original value of 3.96 m to a new one of 5.50 m).

b) The steam relief was automated in order to improve core cooling through the steam generators.

c) A reactor trip signal was added for the case of exceeding a maximum value of reactor building pressure allowed during a LOCA. This signal acts when a loss of coolant occurs at the pressurizer dome, in the safety valve piping or during a failure with such valves at the open position.

d) It was verified that the component vent system (process system) already introduced, is qualified to release to the atmosphere the non-condensable gases from the pressure vessel.

e) The use of a graph indicating "departure from saturation coolant temperature" was implemented for every pressure and temperature values of the primary coolant.

f) It was verified that the manual valves of the main feed-water lines produced a position advise signal at the control room.

g) A manual trip signal was installed for small LOCA's.

No modifications were carried out at CNA-I or CNE as a consequence of Chernobyl accident, due to the design differences between the two plants and the

RBMK reactor. However, the accident had effects on both the operating organisation of the installations and the Responsible Organisation, clearly showing the importance of the compliance with operating procedures, the priority to safety, the personnel training and all the other features that are part of the safety culture concept.

19.6 FIRE PROTECTION

Standard AR. 3.2.3 establishes the safety criteria against fire (or events generated by it) and explosions resulting from fire, that may affect a nuclear power plant radiological or nuclear safety. These criteria include the stages of design, commissioning and operation of the installations.

The fulfilment of the criteria contained in the above mentioned standard is verified through inspections carried out by inspectors and analysts of the Regulatory Body (see Sections 7.3.3 and 7.3.4).

On the other hand, CNA-I and CNE have specific procedures of fire protection. Such procedures contain the description of fire compartments; the composition, responsibilities and functions of the fire brigade; the detection and alarm systems; the extinction systems; the fire-fighting drills and other aspects related to the fight against fire of permanent application in the nuclear power plants.

19.6.1 ANALYSIS OF FIRE RISK

The probabilistic safety assessment being developed at CNA-I (as well as the one that has recently began at CNE) foresees the core damage evaluation due to an eventual fire that may be initiated at a set of areas of the installation known as fire compartments.

The analysis methodology chosen enables the calculation of a core damage probability associated with each of the fire compartments before mentioned, in the case such fire occurs. This methodology comprises a set of tasks such as:

- Establishment of a general procedure for the fire risk analysis.
- Determination of fire compartments, fire barriers and fire propagation routes (see Section 6.3.2).
- Calculation of fire propagation probability to adjacent fire compartments.
- Preparation of a list of affected components at each fire compartment and calculation of the corresponding failure probability rate due to fire.
- Calculation of the core damage probability associated with each fire compartment where a fire occurs.

The following software tools are used for the execution:

- Fire Database NUREG/CR 4586.
- COMPBURN III A computer code for modelling compartment fires - NUREG/CR 4566.
- Database of Fire Compartments at CNA-I. CNEA C RCN IT 055.

The analysis of the results will enable the determination of the highest risk fire compartments for the installation, and the improvements to be carried out either in components or compartments, in order to reduce their failure probabilities in case of fire.

This methodology implies an identification of fire induced failures of components as well as of the several fire propagation routes; therefore it advantageously substitutes the classic failure analysis of common cause due to fire.

19.7 ACCIDENT MANAGEMENT

The procedures for accident management in both nuclear power plants are presently being elaborated.

In relation with this subject, the possible states for a nuclear power plant had been grouped in five levels according to Table 19.1.

Table 19.1

LEVEL	STATE	ACTIONS
1	Normal operation	Operating Manual
2	Foreseen operational incident	Operating Manual
3	Design basis accident	Operating Manual
4	Accident exceeding the design basis	Accident Management
5	Significant discharge of radioactive material	Emergency Plan

The first three levels are covered by operational measures specified in the Operation Manual. As regards level 4, it is additionally split into three sub-levels, according to Table 19.2:

Table 19.2

SUB - LEVEL	STATE	ACTIONS	OBJECTIVE
1	Undamaged Fuel Elements	Ensure core subcriticality and cooling and maintain containment integrity	Prevent core damage
2	Partially Damaged Core	Preserve reactor core controlled	Preserve primary system integrity
3	Failure in Primary Circuit	Preserve radioactive material inside containment	Reduce radiological consequences to public

For the case sub-level three's goal is not achieved, a shift to level five is indicated and the emergency plan is applied.

CNA-I has an emergency plan to be applied inside and outside the installation (see Section 16.4). As already mentioned there are operational procedures for emergency cases that include:

- *Instructions of the* **E** type - they are focused on determining the failure type and taking actions in case of loss of coolant in the primary circuit (big, small or micro-loss). They also deal with steam generator tube breakage, loss of both water supply and hot steam.
- *Instructions of the* **T** type - They indicate how to proceed in the following cases: quick reactor trigger, boron injection, turbine trigger, simultaneous failure of two pumps of the river water supply and switch to emergency current.

It should also be mentioned that according to the operation license requisites, the Responsible Organisation has a personnel re-training program containing actions to be taken during the occurrence of abnormal events and emergencies in the installations (see Section 11.7). Such actions are based on the knowledge and analysis of operation procedures.

CNE has an Emergency Plan to be applied on-site and off-site (see Section 16.4). On the other hand, as already mentioned, the nuclear power plant has an elaborated set of operational procedures for accidental situations known as Operational Procedures for Abnormal Events. These procedures were developed according to the safety design matrix technique (see Section 14.1.2.2) and the Canadian operational experience accumulated along several years of operation. CNE personnel re-training program includes practices in simulators in which abnormal events are developed and analysed according to Operational Procedures for Abnormal Events. This enables procedure updating, as required by the license.

The same importance has been given in such procedures to diagnosis of events and to operation handling to lead the installation to a safe state.

Level four accident management is not taken into consideration in the Operating Manual. Besides, as this level includes accidents exceeding their design basis, it is only possible to prescribe very general measures such as:

- Primary system depressurisation in the case of accidental sequences with core melting at high pressure.
- Controlled gaseous discharges.
- Steam generator feeding with available water.
- Recovery of the electric power system after black-out.

For emergency situations, the emergency plan for both plants considers the immediate notice to the Safety Advisory Internal Committee, Internal Committee of Emergency Control and other organisations (i.e. the Regulatory Body).

In case the accident evolution is such that there is enough time to face level four, the notice to the Safety Advisory Internal Committee and the Internal Committee of Emergency Control is foreseen in order to carry out the analysis of the situation and issue the pertinent recommendations about the accident management.

19.8 COMPLIANCE WITH THE OBLIGATIONS IMPOSED BY THE CONVENTION

The information contained in this and other Articles demonstrates that the country complies with the obligations imposed in Chapter 19 of the Convention on Nuclear Safety.