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Initial Calibration Tests on the FN Tandem Accelerator in Nuclear Regulatory Authority, Buenos Aires

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INITIAL CALIBRATION TESTS ON THE FN TANDEM ACCELERATOR IN NUCLEAR REGULATORY AUTHORITY, BUENOS AIRES

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ABSTRACT

At present, the project consists in concluding the setup of the facility. The Installation will be the first laboratory dedicated to the application of the Accelerator Mass Spectrometry technique in Argentina as well as South America, mainly in the nuclear safeguards area. To carry out this issue, it is necessary to complete the tests to optimize the transmission of the ion beam through the whole system characterizing with precision the performance of each component. The job aims to study that performance for light ion beams, in order to extend in the near future, the capability to measure actinides. On this point, a gas recirculating stripper system is under development.

1. Introduction

Nuclear Regulatory Authority (ARN) is a public Institution, created in 1994, dedicated to the control of nuclear activities in Argentina. It has several laboratories dedicated to research and make measurements related to the regulatory activities. Argentina has substantially increased its cooperation and assistance to the International Atomic Energy Agency in the “international safeguards” field. In this frame, the installation of a dedicated AMS facility has been considered of interest by the ARN. The main purpose of the system will be AMS assays in the nuclear safeguards area as well as environmental monitored. It must be pointed out that this technique is a very powerful tool used by many laboratories for several topics, but only in a few places is used in nuclear safeguards issues.

2. The CEMA laboratory

2.1. Present status

The CEMA laboratory (Centro de Espectrometría de Masas con Acelerador) is based on a FN tandem electrostatic accelerator formerly installed at the Tandem Laboratory at McMaster University, Canada. A description of the AMS system has been covered in another progress report [1].

The installation is completely finished, including the air compressed and water cooling systems as well as insulating gas transfer system. Also an air conditioning enclosure area is included to contain the facility itself. The vacuum value in the whole system is between 1 and 5 x 10^{-7} Torr, except at the stripper box where the vacuum is about 2 x 10^{-6} Torr because there are no vacuum pumps inside the tank. The tubes conditioning was made up to 6.2 MV with N_2 and CO_2 insulating gas at 10 bar, but the terminal voltage must still be calibrated. Only the foil stripper system is installed. The detector system is under tests as well as the micro-channel plates, which will be used for the time of flight system. A FAST multiparameter data acquisition system will be use during the experiments.

AMS is a very sensitive technique which is capable of tracing long-lived radionuclides at concentrations as low as 10^{-16} using only a few milligrams of samples. To achieve this goal, a certain kind of requirements about the machine performance is essential.

The standard way to use the technique is to make radioisotope measurements relative to the stable isotope of the element under study, therefore the transmission of the beam through the facility must be constant for all isotopes of interest (stable and unstable). It means that the performance of the parameters of the machine, like power supplies involved with the ion source, magnets, lenses, steerers, high voltage at the accelerator terminal, etc., must be extremely stables.

A constant transmission will guarantee repetitive measurements, which are necessary because it is very usual to refer the radioisotope concentration of the sample of interest to a standard sample (concentration well known) and blanks (samples with the radioisotope concentration below of the detection limit) [2].

Carbon and Oxygen beams have been tuned, in a very preliminary way, in December 2002 for the first time at about 3.5 MV through the whole facility, but the transmission through the tank was less than 20% and the current measured at the Faraday cup placed before the detector system was very unstable. After those tests, a very careful study of the performance of each component of the system was initiated.

First analyses were made on the injection line consisting basically in changes of position of some components. Later a systematic study of the beam optics, supervised by an expert supported by IAEA, was done. Several tests were carried out and many fails were identified like noisy signals, wrong electrical connections and incorrect disposition of some beam profile monitors. The most important fact discovered was the inadequate response of the new injection magnet due to an error at the cut poles. The consequence associated is that X and Y focus at the magnet are shifted about 500 mm one each other. Nevertheless, after some corrections, a carbon beam was tuned at about 3 MV and 4.4 MV and the transmission through the tank increased up to 50 - 55%.

2.1 Future developments

The next plan is to solve the injector magnet problem installing magnetic singlet quadrupole lens at the entrance and exit of the magnet, which will be donated by the Australian National University (ANU). In order to make measurements with heavy isotopes a gas stripper system should be necessary, instead of a foil stripper. Even excellent foils inevitably change their thickness under bombardment by the beam, producing variations on the transmission through the machine. A modification of the stripper box existing to a gas recirculating system is under development with the priceless assistance of Dr. David Weisser (ANU) and Dr. John Fallon (ANSTO, Australia). The design basically consists in installing an Alcatel turbo pump on the bottom of the stripper box and using a Varian variable leak valve to control the gas flow. Also the replacement of the old electronic devices associated to the terminal steerer plates for Glassman power supplies with double shielding will be done. The foil strippers movement, the variation of the voltage on the steerer plates and the control of the gas valve will be performed by “strings”. For uranium measurements an external stripper will be necessary to get a high enough charge state for the present analyzing magnet to bend it. This stripper could be a foil or a gas one, it is still under discussion.

3. Research program

The ARN is interested in applying AMS technique to measure isotopic ratios in environmental samples taken routinely as well as to detect very low isotope´s activities found in the environment due to anthropogenic nuclear activities taking into account the existing agreements with international Institutions like IAEA.

Once the error at the injector magnet had been solved and with the machine completely characterized it will be feasible to start the measurements with light isotopes like ^{36}Cl in environmental samples. With the gas recirculating stripper system installed it will be possible to measure some heavy isotopes like ^{129}I which is a very interesting isotope. On one hand, since the middle of the last century the environmental levels of ^{129}I with $^{129}I/^{127}I$ ratios of about 10^{-12}

have been dramatically changed as a consequence of the civil and military use of nuclear fission [3]. From radiological point of view, our knowledge about ^{129}I in the environment is still marginal [4]. Although the number of ^{129}I investigations increased dramatically in recent years, systematic studios of this radioisotope should be necessary to close some gaps in our understanding about the environmental abundances of ^{129}I, about its radioecological behavior and to exploit its potential for the retrospective dosimetry of ^{131}I exposure after a nuclear accident [3,5]. Though the environmental ^{129}I levels are not of radiological concern at present, the future development should be carefully monitored. Generally, ^{129}I together with ^{14}C and ^{36}Cl, can be regarded as excellent tools to quantify the long-term impact of nuclear power onto the environment. On the other hand, in our days, concerning to international nuclear safeguards issues, the IAEA has assumed the compromise to verify the correctness and completeness of declarations made by States [6]. The IAEA have been implemented an environmental sampling and analysis. Disturbance of uranium isotopic ratios would be the most evident signature for uranium enrichment. ^{236}U and ^{129}I are important radionuclides for identifying clandestine nuclear activities [7]. Very small samples of environmental material such water, air, soil, vegetables, etc could in principle be used to identify undeclared nuclear sites [8]. With the addition of the external stripper to the facility, uranium measurements will be done. ARN personnel who have already prepared uranium samples for AMS measurements will develop the chemistry sample preparation at ARN laboratories.

4. Summary

The first dedicated AMS facility in Argentina will be operative in the near future to carry out measurements related with routinely environmental sampling and international safeguards issues. In view of the severe impact of environmental radioactivity data on the public opinion and on society in general, proof of quality and reliability is a most important issue and a basic requirement for measurements of environmental radioactivity. This requires a sophisticated system of quality assurance and control.

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Candu Senior Regulators 2005 Annual Report

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CANDU SENIOR REGULATORS
2005 ANNUAL REPORT

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1.- EVENTS:

1.1.- Heavy water spilling through fuelling machine "A":

On June 8th, 2005, Embalse NPP was operating normally at full power after the fuelling machine (F/M) "A" was replaced, and while it was being prepared to operate again, a heavy water spilling within the reactor building occurred. The actuation of the safety system was not demanded and Embalse NPP did not run out of service.

The maintenance activities performed to prepare the F/M "A" included, among others, its calibration, functions check, and heavy water level restitution. When the F/M calibration was concluded, the snout plug was stored in the magazine and the D_2O supply pumps were stopped. Under this condition the involved maintenance personnel required to normalize the F/M "A".

Finally, the F/M "A" was coupled to the new fuel port, the snout plug restored and it was filled up with heavy water again. When this operation had been performed the operator opened the F/M "A" discharge valve (35230 MV37) without lifting the drainage port, and so, when the D_2O supply pumps started, a D_2O spilling into room R 104 occurred.

The main actions taken to control the incidental situation were: to stop the D_2O supply pumps and close both the D_2O feed valves and the corresponding discharge valve. Besides, it was foreseen to improve the operational procedure by giving more relevance to automatic operation.

The incident was attributed to a human error during F/M "A" operation that enabled the spilling of 113 kg of D_2O to the R104 room floor.

2.- REGULATORY SAFETY INDICATORS PROGRAM:

The safety indicators evaluation criteria have been modified according to the experience in applying the regulatory safety indicators program. Owing to difficulties in defining an optimum acceptable or unacceptable value, it was decided to use only one satisfactory zone for the corresponding evaluation criteria, considered as a normal behaviour range for each safety indicator.

The criteria used to define the above mentioned satisfactory zone limits are the following:

- The limit of a satisfactory zone is the most probable range of the frequency distribution (if it is an acceptable value from the regulatory point of view).
- If an indicator has had periods with different frequency distributions, the most conservative distribution is considered.
- If an indicator is strongly dependent on operational conditions, the present condition is considered.
- Limits could be modified according to indicators behaviour.

Three aspects are considered to evaluate safety indicators:

- Comparison of the indicator value with the satisfactory zone
- Indicator trend over the last three-quarters.
- Additional information related to each safety indicator.

3.- PROBABILISTIC SAFETY ASSESSMENT:

3.1.- Shutdown PSA:

The shutdown PSA is still delayed as compared to the required schedule. At present it is going on and is expected to be finished by the end of 2005. At present, there are some still pending tasks. Shutdown PSA includes the following operational states:

- Low power and hot shutdown,
- Power decrease,
- Cold shutdown with open Primary Heat Transport System (PHTS) (Steam Generators - SG - open and SLARETTE equipment operating in the pressure tubes -PTs-),
- Cold shutdown with closed PHTS (SGs closed),
- Cold shutdown with closed PHTS and available service water system (at the last stage, when the Emergency Water Supply –EWS- is connected to the service water system).
- Power start up.

3.2.- PSA – Fire Hazard Analysis:

Fire hazard PSA is still delayed. Up to now a fire generation frequency analysis has been performed as well as its propagation including the affected systems. Impact on systems is still pending, that is to say, results obtained in the above-mentioned analysis still remain to be included in the event trees. This activity will be restarted during 2006.

3.3.- Comparison between Embalse NPP PSA level 1 results and CANDU-6 Generic PSA:

As part of the analysis of Embalse NPP Probabilistic Safety Assessment level 1 (EPSA), the Regulatory Body is carrying out a comparison between the EPSA results and the CANDU-6 Generic PSA (GPSA) that is not finished yet.

This comparative analysis took into account the scope considered in the EPSA, that is, full power conditions, core radioactive inventory, PHTS and moderator system, and the coupled fuel machine. In what follows, differences obtained till now in the comparison are detailed (those items with similar treatment in both PSA are not included).

EPSA and GPSA are supported by different reference documents. For EPSA, the main reference document is “Procedures for Conducting Probabilistic Safety Assessments of Nuclear Power Plants (Level 1), IAEA, SS Nº 50-P-4”. GPSA follows the recommendations contained in the document “USNRC, 1983, PRA Procedures Guide - NUREG/CR-2300: A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants, USNRC Report, NUREG/CR-2300, Volumes 1 and 2”.

3.3.1.- Initiating Events:

Operative experience and reference to other similar PSA analyses were mainly used in the identification and grouping of initiating events (IEs) in the EPSA. In the case of GPSA the Master Logic Diagram (MLD) technique was mainly used.

-. Differences due to Embalse inherent design characteristics

- The IEs group represented by “Heat Transport Leaks Involving Containment Bypass” (IE-LKHX) in GPSA, is considered in EPSA in the following two groups: “primary coolant discharge to the lake through the service water system" (VC) and “primary coolant

discharge to the service building through the service water system" (VD). This is so because GPSA considers the recirculating cooling water (RCW) system, that does not exist in Embalse NPP.

- The "Loss of Condensate Flow to Deaerator" (IE-LOCD) IEs group , considered in GPSA is not applicable to Embalse NPP because of design differences between both NPPs.

- At present are still being analyzed different IEs grouping criteria used in EPSA and GPSA.

3.3.2.- Event Trees:

The most relevant event trees (ETs) were selected on the basis of their contribution to core damage frequency (CDF) in EPSA and they were compared with the corresponding GPSA trees.

Up to now the ETs compared correspond to small LOCA and total loss of service water IEs groups, which are included in both EPSA and GPSA.

EPSA considers the IEs group "Very Small LOCA" (S3) which present the features of exceeding feed pump capability and do not provoke the emergency core cooling system (ECCS) trip. It is important to point out that S3 is the most important IEs group in what concerns to CDF contribution (50%) and it was not considered in GPSA.

3.3.3.- Human actions modelling and data treatment assessment:

- Human actions modelling:

As a result of the comparison between EPSA and GPSA no differences have been found in treatment and development of "Human Actions Modelling and Analysis", as both documents use the Accident Sequence Evaluation Program (ASEP) methodology for its classification and analysis. This tool is a simplified and evolved version of "Technique for Human Error Rate Prediction (THERP)" methodology.

Not only in EPSA but also in GPSA, human actions (HAs) have been typified in pre and post accidental actions. For analysis and modelling of such HAs, EPSA used the Systematic Human Action Reliability Procedure (SHARP) methodology and pre and post accidental HAs as well as human actions diagnosis have been considered.

In the ETs modelling, EPSA included as "Human headers" those post accidental operation team HAs which are necessary to lead the plant to a safe state, tending to improve the accident conditions by using specific plant procedures (diagnosis and execution tasks). GPSA contains no detailed information concerning the HAs included in ETs.

In the fault trees (FTs) modelling , EPSA included the before mentioned pre and post accidental HAs together with the safety systems and the back up systems. GPSA contains no detailed information concerning the HAs included in FTs.

- Data Treatment:

For the quantification process of those human errors identified in the FTs and ETs, according to the HAs type and the model used to carry it out, EPSA used specific plant data, as well as data from the Nuclear Regulatory Commission (NRC) and from the Electric Power Research Institute (EPRI).

Although both EPSA and GPSA have made use of Ontario Hydro´s operative experience as internal data source for the treatment of those data used in HAs modelling, external (international) data sources used in both cases are different.

Data treatment in EPSA has been performed using the PSAPACK v4.3 code, which contains diverse generic data compiled by IAEA and includes experience gained in the nuclear

field. Besides, specific data were extracted from plant event reports corresponding to the period January 1991 / December 2000.

GPSA uses component reliability generic data compiled from CANDU plants operation, mainly from Ontario Hydro NPP, for both external and internal sources. As regards external sources, the main documents used are: “IEEE Standard 500-1984” (failure rates for electric components and I&C) and “Nuclear Plant Reliability Data System (NPRDS) 1983 - Annual Report of Cumulative System and Component Reliability” (nuclear power plants operative experience data collected along 8 years). In the case of internal sources operative experience was used.

4.- SAFETY CULTURE:

Principles and priorities established in previous CSR Annual Reports about general safety principles, safety policy, safety culture (SC), management attitudes, personnel motivation, regulatory activities and voluntary actions of safety improvements and good practices, remain unchanged.

In addition to activities included in the last report, related to the development of an indirect SC indicators set, based on the experience in applying the regulatory safety indicators program, the utility SC assessment using indirect indicators has been carried out. Therefore, the Regulatory Body began to use the following performance indicators, which were already in use, as indirect SC indicators:

- Organisation Related:

  - Operating Experience Feedback Programme.
  - Training.
  - Internal Audits.

- Risk Related :

  - Number of relevant events (*)
  - Safety Systems actuation (**)

  (*) When direct or root causes are associated with human deficiencies.
  (**) Considering here only those related with human failures.

Besides, as a complement to the above mentioned performance indicators, regulatory audit results are also used as a S.C indirect indicator.

Concerning safety culture in the Regulatory Body, a specific regulators training course on Human Performance to improve their qualification to deal with utility organisational factors was performed.

5.- DOSES AND ALARA PROGRAMMES (EMBALSE NUCLEAR POWER PLANT):

5.1.- Radioactive releases into the environment:

The Regulatory Body authorised a set of gaseous and liquid discharge limits, contained in the plant operating license and shown in Tables 1 and 2 respectively. For critical group doses, these limits were set much lower than 0.3 mSv.

Table 1. Authorised gaseous discharge limits for Embalse

NUCLIDE	K_i (TBq)
Ar-41	$7,4 \times 10^{3}$
Kr-85m	$3,7 \times 10^{4}$
Kr-87	$7,4 \times 10^{3}$
Kr-88	$3,7 \times 10^{3}$
Xe-133	$1,9 \times 10^{5}$
Xe-135	$3,7 \times 10^{4}$
H-3	$3,7 \times 10^{4}$
I-131	$2,2 \times 10^{1}$
Co-58	$3,7 \times 10^{1}$
Co-60	$3,7 \times 10^{-1}$
Sr-89	$1,1 \times 10^{2}$
Sr-90	$3,7 \times 10^{0}$
Ru-106	$1,5 \times 10^{0}$
Cs-134	$1,5 \times 10^{0}$
Cs-137	$3,7 \times 10^{-1}$
Ba-140	$1,5 \times 10^{2}$

Table 2. Authorised liquid discharge limits for Embalse

NUCLIDE	K_i (TBq)
H-3	$3,7 \times 10^{3}$
Cr-51	$3,7 \times 10^{2}$
Mn-54	$7,4 \times 10^{-1}$
Fe-59	$3,7 \times 10^{1}$
Co-60	$1,5 \times 10^{-1}$
Zn-65	$7,4 \times 10^{-2}$
Ni-65	$7,4 \times 10^{3}$
Sr-89	$3,7 \times 10^{0}$
Sr-90	$1,5 \times 10^{-1}$
Zr-95	$1,9 \times 10^{0}$
Ru-103	$3,7 \times 10^{0}$
Ru-106	$1,5 \times 10^{-1}$
Ag-110m	$1,1 \times 10^{0}$
Sb-125	$1,1 \times 10^{0}$
I-131	$1,9 \times 10^{-1}$
Cs-134	$3,7 \times 10^{-2}$
Cs-137	$3,7 \times 10^{-2}$
Ba-140	$1,1 \times 10^{1}$
Ce-144	$1,9 \times 10^{-1}$
Gd-153	$3,0 \times 10^{1}$

The gaseous radioactive releases from Embalse NPP to the environment, for the period 2002-2004 is shown in Table 3, discriminating those corresponding to I-131, H-3, aerosols and noble gases, and including an estimation of C-14 discharges.

Table 3 - Activity released from Embalse to the environment as gaseous discharges

Year	I -131 (TBq)	Tritium (TBq)	Aerosols (TBq)	Nobles Gases (TBq)	C -14 (TBq)
2002	$0,0 \times 10^{0}$	$2,7 \times 10^{2}$	$0,0 \times 10^{0}$	$2,4 \times 10^{1}$	$4,2 \times 10^{-1}$
2003	$0,0 \times 10^{0}$	$2,6 \times 10^{2}$	$0,0 \times 10^{0}$	$7,2 \times 10^{1}$	$4,8 \times 10^{-1}$
2004	$5,3 \times 10^{-7}$	$3,3 \times 10^{2}$	$8,3 \times 10^{-7}$	$4,5 \times 10^{1}$	$4,4 \times 10^{-1}$

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

The liquid radioactive discharges from Embalse to the environment for the same period are presented in Table 4, discriminating between liquid discharges of H-3 and gamma emitters.

Table 4 - Activity released from Embalse to the environment as liquid discharges

Year	Tritium (TBq)	Other radionuclides (TBq)
2002	$6{,}9 \times 10^{1}$	$1{,}6 \times 10^{-3}$
2003	$1{,}1 \times 10^{2}$	$1{,}8 \times 10^{-3}$
2004	$8{,}3 \times 10^{1}$	$1{,}9 \times 10^{-3}$

The 89% of the total average discharge from Embalse to the environment corresponds to tritium. These average discharges were less than 3% of the annual authorised discharge limits.

5.2.- Public exposure:

The annual critical group doses due to Embalse operation during the period 2002-2004 are presented in Table 5, and discriminated according to discharge type. These values resulted lower than 1% of the established individual dose constraint. Liquid discharges were the main contributors.

Table 5- Critical group individual dose for Embalse

Year	Gaseous discharge doses (mSv)	Liquid discharge doses (mSv)	Total doses (mSv)
2002	$1{,}4 \times 10^{-4}$	$1{,}7 \times 10^{-3}$	$1{,}8 \times 10^{-3}$
2003	$1{,}9 \times 10^{-4}$	$2{,}3 \times 10^{-3}$	$2{,}5 \times 10^{-3}$
2004	$1{,}9 \times 10^{-4}$	$2{,}3 \times 10^{-3}$	$2{,}5 \times 10^{-3}$

The regional collective effective doses, calculated with population data up to a radius of 2000 km from the nuclear power plant, normalised per unit of electric energy generated, are presented in Table 6.

Table 6 - Regional normalised collective effective dose for Embalse

Year	Gaseous discharge doses (manSv/GWa)	Liquid discharge doses (manSv/GWa)	Total doses (manSv/GWa)
2002	$1{,}5 \times 10^{-2}$	$7{,}4 \times 10^{-2}$	$8{,}9 \times 10^{-2}$
2003	$1{,}4 \times 10^{-2}$	$9{,}9 \times 10^{-2}$	$1{,}1 \times 10^{-1}$
2004	$1{,}8 \times 10^{-2}$	$8{,}3 \times 10^{-2}$	$1{,}0 \times 10^{-1}$

The average regional collective effective dose per unit of electric energy generated, for the period 2002-2004, was 0,1 man $Sv/GW_{(e)}a$.

For the same period (2002-2004) the average global collective effective dose per unit of electric energy generated was 0,1 man $Sv/GW_{(e)}a$ for tritium discharges and 18 man $Sv/GW_{(e)}a$ for the estimated C-14 releases to the environment. These global collective effective doses correspond to the truncate effective dose commitment integrated over the expected duration of the practice (500 years).

5.3.- Occupational dose in Embalse nuclear power plant:

The collective effective doses, the normalised collective effective doses and the average individual effective doses received by Embalse nuclear power plant workers during the period 2002-2004 are presented in Table 7.

Table 7 - Occupational dose in Embalse

Year	Collective effective doses (manSv)	Normalized collective effective doses (manSv/GWa)	Average effective doses (mSv)
2002	4,6	9	5,0
2003	1,3	2	1,9
2004	3,0	5	3,2

During 2004, Embalse NPP performed the programmed maintenance outage, which is associated with the increase in the annual collective dose. No worker exceeded the annual dose limits.

6.- CONVENTION ON NUCLEAR SAFETY:

The third National Nuclear Safety Report (NSR) was presented at the review meeting of the Nuclear Safety Convention held in April 2005 at Vienna. The NSR included all safety aspects of the Argentinean nuclear power plants and the measures taken to enhance the safety of the plants.

During the NSR presentation Argentina demonstrated that all the pending issues identified during the second review meeting discussions (related to emergency preparedness, radiation dose levels at Atucha I NPP and safety performance indicators) were satisfied according to the Nuclear Safety Convention requirements.

As a general conclusion it can be stated that Argentina fulfils all the obligations imposed by the Nuclear Safety Convention.

The questions asked to the Argentinean delegation before the meeting and during the presentation concerning specifically to PHWRs were mainly addressed to the following issues: pressure tubes, feeders, emergency preparedness, Atucha I NPP back-fitting program, quality management system in the regulatory body and Atucha II NPP. The main issues arising from the review meeting were the following:

6.1.- Pressure tubes:

The SLARADE computer code was used to define the repositioning strategy, incorporating all the operating experience of the Canadian plants, so that a better estimation about the time window for contact between pressure tubes and calandria tubes was achieved.

In the last programmed outage a pressure tube campaign was conducted and ten tubes were selected for scraping, three of them also scraped in 1998. The samples were analysed at AECL laboratories in Canada. Deuterium pickup was determined for five positions in each of the tubes considered and raw data is available.

6.2.- Feeders:

During each planned outage several feeder areas were inspected, with the purpose of determining the thickness wall decrease by erosion – corrosion mechanisms and cracking measurements. Inspection methods and qualified personnel used are in accordance with standards and guides. The results obtained in the inspection showed that life expectancy was higher than 25 years for all sensitive areas. In the case of flow accelerated corrosion (erosion-corrosion) damage, ultrasonic inspection results showed that the feeder average thickness was similar to that obtained in previous inspections. Therefore the thickness-decreasing rate is practically constant showing that there is no damage acceleration mechanism.

Similar conclusions were stated in the case of stress corrosion cracking. Crack detecting and sizing methodology was used (according to COG procedure) and it could be concluded that there are no relevant indications in whole feeders.

6.3.- Emergency Preparedness

The Emergency Plans for both NPPs in operation (Atucha-I and Embalse NPPs) have dedicated chapters to the consideration of off-site and on-site aspects. Off-site aspects are under the Regulatory Body responsibility, regarding radiation protection of the public and the involved Emergency Organisations. The Responsible Organisation acts on behalf of the Regulatory Body during the first hours from the beginning of the accident until the emergency control is taken by the Regulatory Body and during this period it is responsible for the application of the first protection measures. On-site aspects are always under the Responsible Organisation responsibility.

According to the Operation License, an annual emergency exercise is carried out at each NPP. On-site emergency exercises are performed to test the capability of the whole plant emergency organisation to mitigate the accident and to minimise its radiological consequences. Additionally, abnormal event procedures are used for training in Atucha I and Embalse NPPs at the full scope simulators.

In the on-site exercises, postulated event sequences that lead to core damage and radionuclide release are defined. During the accident progression different plant response groups take part. Initially, the reactor operation team follows the emergency procedures and it requires the action of plant response groups and reports to external organisations. Later, several actions like sheltering, evacuation, and also thyroid prophylactics of plant workers (simulating the use of potassium iodide -IK- pills) are implemented. A radiological control of the plant staff is performed during the exercise.

Once the emergency exercise is concluded wrap-ups and discussions to consider the different findings are carried out and a report including strengths and weakness is issued. Finally, corrective actions are taken, if necessary.

The Regulatory Body carries out a close follow-up of the different emergency exercise stages, listing findings independently of the Responsible Organisation.

6.4.- Atucha I NPP Back-fitting Program:

The Regulatory Body required a back-fitting program giving priority to reactor internals issues. The program also considered relevant safety aspects such as completion of the second heat sink system and the reactor pressure vessel analysis. Besides, an updated version of the relevant safety documentation was required. All the back-fitting activities were fulfilled in accordance with what the Regulatory Organisation required. The main activities performed were:

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

6.5.- Quality Management System in the Regulatory Body:

In the past, some activities were performed in the area of quality management within the Regulatory Body. The Board of Directors decided to stress a plan for the development and implementation of such activities during 2002 that should include the entire institution. Although quality activities were usually considered in the ARN, it was decided to work in this area in a

systematic way with the support of external advisors belonging to the National Technological University of Buenos Aires (UTN-BA). With this purpose, an agreement was signed between ARN and UTN-BA. Its goal is to obtain a well-structured quality management within the ARN taking into account external and internal customers, and based on the "Continuous Improvement Concept". ISO 9000, version 2000 and IAEA document PDRP-6 "Quality Management on the regulatory bodies" were the main support documents.

The quality management implementation and control area is an independent unit within the structure and will depend of the Board of Directors. Three branches compose this unit: Internal Audits, Continuous Improvement and Documentation Control.

Specific training of different staff positions was based on ISO standards, IAEA document PDRP-6, quality system tools, working groups, continuous improvement, customer services, group of facilitators, internal communication, workshop on performance indicators and documentation control and classification.

The Regulatory Body quality management program represents a great effort of the staff, and particularly of the high level staff.

6.6.- Atucha II NPP construction:

During 2004, the Responsible Organisation resumed the negotiation with the main contractor –Siemens- in order to discuss technical and financial conditions and establishing a program to continue with the construction and finalise the plant.

On the other hand, the Regulatory Body prepared a project review considering present standards and those in force when the plant was designed, concluding that it is necessary to perform a design review according to the State of the Art, prior to finish the plant construction.

The Construction license still in force was also reviewed, and some modifications could be necessary to finalise the plant. In addition, the Regulatory Body is analysing the convenience of requesting IAEA a design review mission. Such mission would be useful taking into account that the plant was designed 30 years ago and it has been a long and interrupted construction period.

7.- ANNULUS GAS SYSTEM LEAK:

The Annulus gas system (AGS) of the Embalse NPP is basically composed of carbon dioxide (CO_2) circulating in the annulus gap existing between the pressure tubes (PTs) and the calandria tubes (CTs). The AGS function is to provide thermal isolation between PTs and CTs, to detect and locate PTs leakages and to prevent fuel channel corrosion by means of a carbon dioxide (CO_2) atmosphere. The normal AGS operation range is defined by the variation of the CO_2 dew point temperature, between -35ºC and -10ºC. When the value of -10ºC is reached, it is necessary to purge the AGS. If the dew point temperature can not be reduced, in such case, it should be assumed that a relevant loss of coolant has occurred.

Historically, the frequency of AGS cycle purging ranged between 15 to 7 days. However, it was observed that such frequency began to decrease with time, reaching an intermediate value of 4 / 5 days and then approximately 2 days, which is considered as an AGS water ingress (wet) indicator. The possible sources and causes were analysed. It was concluded that the humidity increase was due to the light water ingress from the End Shield Cooling System (ESCS) through some place located in the lattice tube area corresponding to one or more of the fuel channels C08, E08, G08, J08, L08, N08, P08, R08, T08 and V08, belonging to the same channel column (8 column). Based on the dew point temperature evolution, an average leakage rate of 0.225 H_2O gr. per hour was estimated.

The designer, AECL, was contacted and some corrective actions focused to improve the AGS condition in case of a leakage increase were taken. These corrective actions included:

- To divide the AGS into two parallel and independent loops (isolated loops): one of them including the 10 fuel channels belonging to the affected column and the other one the remaining 370 channels;
- To implement a continuous “sweep” flow toward the atmosphere in that loop including the fuel channels belonging to the affected (continuously purged) column and
- To identify the affected fuel channel.

Since the beginning of the implementation of two isolated loops, the AGS operated mainly in a continuous purged mode enabling to maintain the required dew point temperature.

Finally, the V08 fuel channel was identified as the location of H_2O ingress from the ESCS to the AGS. Besides, it was assumed that the end shield coolant was making its way into the annulus gas system via a flaw in a lattice tube weld.

The V08 fuel channel is located in the last position of the 8 column, this means that V08 is the last channel in the gas circulation direction before being discharged to the atmosphere. Therefore, considering that the AGS is mainly operated in the continuous purged mode, the humidity from channel V08 is directly discharged to the reactor building environment, thus preventing humidity from returning to the other 9 channels. This means that humidity could only affect channel V08.

The analysis and evaluations performed by the utility and the designer enabled to conclude that until 2010 it will not be necessary to identify the affected location and remove the corresponding fuel channel in order to repair the flaw through which light water enters the AGS.

The Regulatory Body has evaluated the above mentioned issue and set the following requirement to the utility:

- During the 2004 programmed outage the affected channels must be repositioned, to assure that before the 2005 scheduled outage, there will be no PT/PC contact. If it is not possible, then demonstrate that the presence of light water in the annulus gas will not imply that the PT equivalent hydrogen content reaches the Blister Formation Threshold (BFT) before the 2005 scheduled outage.
- To survey the involved channels continuously.

The utility complied with the above mentioned requirement by repositioning all the 10 channels involved in order to ensure that before the 2005 scheduled outage there will be no PT/PC contact.

On the other hand, the involved channels are continuously surveyed through dew point temperature measurements and determination of water characteristics. Besides, for each scheduled outage, an ISI to the involved channels is programmed.

8.- IRRADIATED FUEL ELEMENT DRY STORAGE SYSTEM:

The utility has programmed to expand the Irradiated Fuel Element Dry Storage System capacity by constructing new silos to satisfy the plant operation needs for the next five years. The construction of sixty-four new silos was programmed for the period 2005 - 2006 (twelve months) and it began in last July.

The new silo capacity consists in nine baskets with 60 fuel elements per basket, which means a storage capacity of 540 fuel elements per silo. Therefore, the total storage capacity for the 64 new silos will be 34560 fuel elements.

Design, construction and assembly of the new silos do not foresee changes from the existing silos, which means that safety related issues fulfil requirements as well as the existing Irradiated Fuel Elements Dry Storage System, already licensed.

In order to licence the above-mentioned new silos, the Regulatory Body has implemented a construction and assembly follow up program based on inspections to the different activities. Moreover, when these activities finish, a behaviour analysis together with field measurements will be required in order to compare calculated with actual values.

9.- PLANT LIFE MANAGEMENT:

It was decided to implement a Plant Life Management Program (PLIM) in Embalse Nuclear Power Plant (ENPP), with the purpose of reaching its design life ensuring safe operation, providing an adequate tool for decision making and establishing the basis for a possible life extension.

Since the end of 2004, interaction with IAEA was initiated with the purpose of obtaining support for the implementation of PLIM at the ENPP. The corresponding agreement was approved and it is now at an administrative stage in order to be authorised as a Human Resources Program.

During February 2005 a group of selected personnel coming from CNE as well as from the National Atomic Energy Commission (CNEA), were trained by the designer Atomic Energy Canadian Limited (AECL) in applicable methodologies for the implementation of PLIM. At the moment and during a first stage, which is estimated will be finished at the end of 2005, a pilot analysis is being performed that includes some relevant components and, when it is completed it will be evaluated together with AECL specialists.

The PLIM for ENPP is mainly a program for the analysis and management of Systems, Structures and Components (SSC) ageing, which enables the safe operation of the installation during an established period including not only its life-time as foreseen during the design stage, but also an eventual life extension. This means that those activities included in PLIM are precursors of refurbishing activities that would be necessary in order to decide a plant life extension.

During PLIM implementation it is foreseen to perform a control, updating and improvement of operation, maintenance, modification and inspection programs trending to increase its ageing management efficiency. In order to achieve these goals, PLIM requires complementing and linking the above-mentioned programs in a systematic way, so as to minimise the impact on safety due to SSC ageing.

PLIM is similar to the program used in other CANDU NPPs and it is constituted by three stages:

Stage 1: analysis of critical SSCs (SSCC) by means of condition assessment (CA), life assessment (LA), systematic assessment of maintenance (SAM) and integrated analysis of safety and performance;

Stage 2: implementation of the assessment results obtained during stage 1 that will enable to reach design life time under safe and economically viable conditions and;

Stage 3: Plant Life Extension plan (PLEX), that includes the refurbishment and extended plant operation.

As already mentioned, ENPP is now implementing the PLIM Stage 1 as a pilot test. Depending on the results obtained during this pilot test, it is foreseen to continue with Stage 1 until the end of 2008. At the same time, it is foreseen to begin during 2006 with the implementation of Stages 2 and 3 that will be extended to 2010 and until the end of the eventual extended life time respectively.

It is important to point out that in PLIM stages 1 and 2, life extension is postulated as a possibility. During PLIM stage 3 technical and economical studies will be carried out in order to

support an eventual life extension request, the associated necessary engineering tasks and the refurbishment implementation to ensure ENPP extended operation under safe conditions.

The SSCC are selected by using a screening analysis based on degradation mechanisms associated to materials and on operational experience. The SSCC involved in stage 1 pilot test are the following:

a) Steam generators;
b) feeders;
c) feed pumps;
d) feed-water pumps;
e) moderator heat exchangers and,
f) shutdown cooling system heat exchanger.

As a result of stage 1 pilot experience, it is expected to predict the SSCC behaviour, that will enable to propose adequate recommendations to implement the necessary corrective, preventive and predictive actions, as well as to estimate SSCC life time and, eventually, the corresponding extended life time.

It should be pointed out that the utility has been implementing, though not systematically, a set of actions related to PLIM program. As a result, the analysis of life assessment for steam generators, pressure tubes and feeders is almost complete. Nevertheless, a steam generator remaining life estimation is still pending, task foreseen when the implementation of measures to mitigate degradation of critical mechanisms be finished.

The Regulatory Body did not require PLIM implementation for ENPP. Nevertheless, in 1998, it required the utility to implement an Ageing Management Program, in force since then, which includes the following:

- Selection of the most important components (SSCC) for plant safety where the ageing effects must be evaluated;
- Study of the selected SSCC ageing mechanisms and identification and/or development of monitoring practical methods and;
- Assessment of remaining life of the selected SSCC and management of degradation due to ageing, through surveillance, maintenance and operation practices.

The Regulatory Body has also decided to carry out a close follow up of all those activities involved in the PLIM program, in order to collect the necessary information and knowledge required to be able to make a decision as refers to an eventual ENPP life extension.

10.- PRESSURE TUBES SCRAPE SAMPLING:

A pressure tubes (PTs) scrape sampling was performed at Embalse NPP during the last planned outage. According to Canadian Standard CAN / CSA N285.4-94, samples of pressure tubes were extracted with the purpose to monitor the deuterium concentration. Samples were taken from the pressure tubes inside surface, in different axial locations in order to provide an indication of variation in deuterium concentration along the length of the PTs.

The PTs to scrape were selected by using a criteria based on the necessity to have a whole core deuterium concentration description. Also, PTs were selected in addition to those scraped during the first Embalse scrapping campaign performed in 1998. A total of seventeen PTs were visited in 1998 and 2004 campaigns. Ten PTs were visited in each scrape campaign. Three PTs were sampled in both scrape campaigns in order to investigate the possibility of an increasing deuterium uptake rate (DUR). The technique used was “wet scraping” cutting the sample fully immersed in heavy water coolant under maintenance flow condition.

The analysis of the samples were done using two different techniques: Differential Scanning Calorimetry, DSC (that yields hydrogen equivalent concentration) and Hot Vacuum Extraction

Mass Spectrometry, HVEMS (from which is possible to know both the hydrogen and deuterium contents). DSC technique was used only for supporting HVEMS results.

The analysis results concluded that values of Embalse DUR are lower than in other CANDU plants of similar age than Embalse. According to what was determined in the 2004 campaign, deuterium concentration ranges from 4,7 ppm to 39 ppm and, considering that in the 1998 campaign this range was 3,1 ppm to 26 ppm., it was concluded that the deuterium concentration was increased approximately 50 % in 5,5 years. The DUR of the fuel channels re-scraped, at 4m location (channels M12, O02 and S17), were increased since the 1998 scrape sampling. However, the average DUR values at the outlet end of pressure tubes (at 5,2 m location) corresponding to the location where blister susceptibility was calculated, has not increased.

The DUR results will be used together with the deformation analysis of the whole core from SLARADE code to define the PTs repositioning strategy to be implemented in next planned outages. Therefore, considering the above mentioned, at present it is being evaluated when the next scrape campaign will be performed which, according to designer recommendations, would not exceed three years from the last campaign.

The Regulatory Body requested to the utility that PTs garter spring repositioning be performed as a mean to ensure that contact between PTs and calandria tubes (CTs) is avoided in such PTs that would have reached the blister threshold formation (BTF).

11.- FEEDERS:

According to Canadian standard CNA/CSA.285-4/94, and the plant document “In Service Inspection Program PI-1073 - Appendix A”, measures of wall thickness were carried out in the outlet feeder tubes of Embalse NPP in order to determine the evolution of its wall thinning.

The feeders wall thinning is mainly produced by flow assisted corrosion (FAC), biphasic flow, temperature, primary coolant physical-chemical parameters (like Pd, conductivity, oxygen content, etc) and, material composition (mainly chromium content).

Therefore, the feeder wall thickness measures were carried out only in the outlet feeders because wall thinning due to the above mentioned effects is more important due to the existence of biphasic flow. The measures were carried out in the area where the wall thinning is great: in the extrados area of the first bend, near the Gray-lock. The obtained data are used to monitor the remaining life of the feeder tubes.

At present, 379 out of 380 outlet feeders has been measured, according to the following detail:

1. Side A
   - 97% of the outlet feeder bends of ∅2” (29 bends; the outlet first bend of channel O-22 remains without being inspected due to access difficulties)
   - 100% of outlet feeder bends of ∅2.5” (160 bends).

2 Side C
   - 100% of outlet feeder bends of ∅2” (30 bends)
   - 100% of outlet feeder bends of ∅2.5” (160 bends)

The above mentioned wall thickness measures enable to conclude that all the feeders would reach the design life with wall thickness greater than the minimum thickness allowed for normal operation. However, for a group of 39 bends critical feeders it was estimated that the minimum wall thickness would reach values close to the acceptable minimum by the end of design life and, for this reason, it will be necessary to monitor them to ensure that they are all fit for service. The design life was established in 24 Equivalent Full Power Year (EFPY), and the minimum allowable wall thickness is: 3,25 mm for ∅2,5” and 2,67 mm for ∅2” feeder bends.

On the other hand, an ultrasonic examination for feeder axial crack detection was also performed. The examination procedure was intended to detect and verify the existence of axial

inner diameter flaws in outlet feeder bends due to the process of cold bending work or stress corrosion cracking phenomenon.

Ultrasonic examination, which enabled to detect sizes up to 1mm. of thermal fatigue cracks, was performed to all the feeders bends (2" and 2,5"). The results of the inspection showed the non-existence of any relevant indications.

The Regulatory Body carried out a follow up of the feeders condition evolution through the analysis of data obtained and verifying compliance with the Canadian Standard CNA/CSA.285-4/94 as well as with the Embalse NPP In Service Inspection Program.

12.- BIVALVE MOLLUSCS:

During the scheduled outage carried out in 2004, the presence of an important quantity of bivalve molluscs (limnoperna fortunei) was detected in the first 200 meters of the discharge channel, as well as inside pipes, pumps, valves and heat exchangers.

Since then, the problems produced by such molluscs increased, generating partial obstructions in the process water system piping that caused temperature increases and flow decreases.

The Regulatory Body required to the utility an assessment of the impact on safety produced by the presence of molluscs in different components of the installation systems as well as a definition of the foreseen actions in order to face this problem.

In order to comply with the above-mentioned requirement, the utility carried out a probabilistic assessment of the impact on safety produced by the presence of bivalve molluscs. At present, the Regulatory Body is still evaluating the assessment. As a result, the utility informed that they do not affect the plant safety through the heat exchangers that are monitored on line during plant normal operation. On the other hand, it was verified that the molluscs could affect heat exchangers which remains in stand by during normal operation.

It was also concluded that some of the affected systems are: Emergency Core Cooling System (ECCS), Emergency Water Supply System (EWS) and Process Water System. It was particularly observed that both ECCS (3432-HX1) and EWS (3341-HX1 / 2) heat exchangers were the most affected.

The actions taken by the utility in order to comply with the regulatory requirements for each case are listed as follows:

<u>ECCS heat exchanger:</u>

The ECCS heat exchanger is, during normal operation, in a stand-by condition and it was considered in the PSA as a "non-repairable component in operation", which means that the only failures considered in the model were those that could occur during its functioning time and after the occurrence of an initiating event. It was thus assumed that it was not possible for the heat exchanger to be obstructed under a stand-by condition. In this case the corresponding core damage frequency (CDF) was 2,67 E-5 / year. However, considering the bivalve molluscs effect, it was necessary to consider a "stand-by tested component" reliability model, which results in a CDF value of 7,15 E-5 / year (without considering its failure rate change due to bivalve molluscs effect).

The PSA results showed a great sensitivity to the reliability model used, due to the fact that the heat exchanger has a significant role in loss of coolant events, it is a non-continuously monitored passive component and its test frequency is once in six years. Besides, it was determined that the CDF is strongly dependent on the test frequency of the ECCS heat exchanger, so that for a test frequency of one month the CDF does not change as compared with the CDF previously calculated in the PSA – Full Power, being its value of 2.46 E-5 / year. Consequently, a repetitive monthly test for the heat exchanger secondary side flow measurement was implemented.

EWS heat exchanger:

The total non-availability of the EWS was analysed taking into account that this system, although weekly tested, has a high probability of being obstructed by the presence of the bivalve molluscs. The results of such analysis show that the CDF varies from a value of 2,46 E10-5 / year (EWS available) to 9.96 E10-5 / year (EWS non-available).

Regarding the relevant CDF increase, the possibility of including a periodic visual inspection in the routine tests during the most molluscs polluting months is at present under consideration, particularly for the first meters of the lake water suction pipes.

Corrective actions:

In order to reduce the population of molluscs larvaes present in the installation cooling systems, the utility decided to clean the most affected components and dose chlorine into the water pumped from the lake. Besides, an agreement was established with the Buenos Aires University to study the development of bivalve molluscs and to optimise the water chlorine concentration. Such studies are not finished yet.

13.- QUALITY MANAGEMENT SYSTEM IN THE REGULATORY BODY:

The ARN decided to implement a quality management system (QMS) with the support of external advisors who belong to the National Technological University of Buenos Aires (UTN-BA). An agreement was then signed between the Regulatory Body and UTN-BA with the purpose of obtaining a well-structured quality management within the ARN focusing external and internal customers and based on the “Continuous Improvement Concept”. ISO 9001, version 2000 “Quality management system requirements” and IAEA document PDRP-6 “Quality Management on the regulatory bodies” were the main support documents.

The QMS has been implemented by an independent unit (QM unit) within the organisation. It is responsible of managing the Regulatory Body quality system in search of continuous improvement of the satisfaction of the internal customer as well as those groups that interact with the institution. It depends of the Board of Directors, and has three branches: Internal Audits, Continuous Improvement and Documentation Control.

The functions of this unit are:

- Carry out a centralised control of the QMS documentation;
- Carry out and implement the QMS audits annual program and;
- Facilitate the Regulatory Body continuous improvement tasks, according to strategic needs, to audit results and the evolution of international standards.

By the end of 2004 the QMS in the Regulatory Body was reinforced, by widening the knowledge about quality principles and managing the organisation as a coherent set of processes. It was thus necessary to name process responsible persons, to develop a wide process matrix (mapping) and the relationship between them.

In 2005 working plan, the following projects (quality sub-systems) still at a developing stage, have experienced some advance:

- Accreditation of Regulatory Body laboratories according to what is established in ISO / IEC 17025 Standard – 2005 version “General requirements for testing and calibration laboratories competence" and;

- Certification of the process of radioactive material safe transport assessment as established in ISO 9001:2000 Standard.

- Certification of Regulatory Body (ARN) - Buenos Aires University (UBA) -International Atomic Energy Agency (IAEA) joint post-grade course according to what is established in IRAM 30000 Guide – “Guidelines on the Interpretation of ISO 9001 Standard for Education”;

Compliance with planning related to QM is delayed because the release of applicable documents was affected by the following:

- The lack of enough human resources and the corresponding increased work demand in the different operative groups delayed the expected response in the system development and;
- In January 2005 some relevant changes in the Regulatory Body organisation were introduced generating, in turn, the needs of modifications to procedures and work instructions already approved, task that is being carried out at present.

Nevertheless, the following activities were carried out related to the QMS:

1) There was an advance in the ending and release of the following QMS documentation:

- Regulatory Body Quality Manual
- Set of procedures and work instructions;

2) A workshop took place on audits with the participation of 17 professionals trained in the QMS internal and external audits. Internal audits are foreseen to begin during 2006.

3) The following accreditation courses based on ISO / IEC 17025 Standard, that included some tests performed in Regulatory Body laboratories were given.

- Quality management and technical competence in testing laboratories;
- Workshop on documentation;
- Metrology and reference materials;
- Measurement uncertainties;
- Validation of test methods and;
- Internal audits.

It is foreseen that the accreditation will include the entire test techniques used in the Regulatory Body laboratories. For the time being, the above-mentioned accreditations began to be carried out to tests performed in the following laboratories: tritium determination in water (direct scintillating method), uranium in water (direct fluorimetric method) and gamma emitter nuclides (spectrometry method).

Until now the experience on implementation such QM system revealed significant institutional changes. Next year it is foreseen to begin a more systematic implementation of the continuous improvement methodology as a result of the strategic goals and internal audits.

14.- ROOM R001 LEAKAGES:

During last September a leak test in room R001 was performed. The results showed the existence of five leakages from R-001 toward R-201, R-103 and R-104 rooms respectively. Besides, a verification of those leakage locations that had been repaired after the former test showed that they remained in good condition without leakages.

However, considering that overall R001 leakage tests showed the existence of leakages in previously repaired locations as well as in new locations, the Regulatory Body is considering the possibility of requiring the utility to perform such tests more frequently.

Método simple para la determinación de ^{210}Pb y ^{226}Ra en aguas

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MÉTODO SIMPLE PARA LA DETERMINACIÓN DE ^{210}Pb Y ^{226}Ra EN AGUAS

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RESUMEN

En este trabajo se describe un método para la separación y posterior determinación de los radionucleidos ^{226}Ra y ^{210}Pb en muestras de aguas ambientales. La determinación de ^{210}Pb consiste en su precipitación como $PbSO_4$ (sulfato de plomo) y posterior medición por centelleo líquido. La determinación de ^{226}Ra se realiza por co-precipitación con $BaSO_4$ (sulfato de bario), emanación de ^{222}Rn en tolueno y su posterior medición por centelleo líquido. Ambos procesos son rápidos, sencillos y utilizan reactivos accesibles y de bajo costo. En el caso de la determinación de ^{210}Pb, se optimizaron los parámetros relacionados con la separación y medición. En el caso de la determinación de ^{226}Ra, se consideró la técnica ya empleada por el laboratorio para la medición de este radionucleido.

INTRODUCCIÓN

La radiactividad natural está presente en las aguas debido a la presencia de elementos radiactivos naturales en la corteza terrestre. Además, ciertas actividades humanas (extracción y procesamiento de la minería del uranio, arenas minerales, industria de los fertilizantes, etc.) pueden incrementar las concentraciones de radionucleidos naturales en el ambiente y por ende en las aguas superficiales[1].

El ^{210}Pb pertenece a la serie de decaimiento del ^{238}U y tiene una vida media de 22,4 años. El gas radón decae en ^{210}Pb a través de varios radionucleidos de vida media corta. Es uno de los radionucleidos naturales más radiotóxico que se encuentra en las aguas de consumo[2]. Cuando el ^{210}Pb es absorbido en el cuerpo humano, se concentra en hueso y es eliminado en forma lenta[3]. En aguas de pozo, donde la concentración de radón puede alcanzar valores importantes, también pueden hallarse por decaimiento radiactivo del gas concentraciones elevadas de ^{210}Pb.

Es de particular interés también la evaluación de los niveles de ^{226}Ra debido a su larga vida media y a su significativa incidencia en la dosis por ingestión[4].

Es por ello que existe la necesidad de desarrollar e implementar métodos para la evaluación de ambos radionucleidos.

Generalmente, la medición de ^{226}Ra se realiza por espectrometría α, espectrometría γ o medición por centelleo líquido. En el caso de la espectrometría α, este método es el más sensible, permitiendo la evaluación de los distintos isótopos de radio, sin embargo el proceso químico requerido no es sencillo[5]. En el caso de la espectrometría γ, la determinación es rápida pero tiene interferencias y su límite de detección es elevado[5]. Es por ello que el método seleccionado por nuestro laboratorio para la determinación de ^{226}Ra se basa en la medición en un equipo de centelleo líquido[6].

La determinación de ^{210}Pb puede hacerse también por espectrometría α, espectrometría γ de baja energía o medición por centelleo líquido[7]. En el primer caso la determinación de la concentración de ^{210}Pb se realiza a través de la medición de ^{210}Po, luego de alcanzado el equilibrio[8]. Es un método sensible, pero se necesita mucho tiempo para obtener el resultado. En el caso de la espectrometría γ[9,10] el límite de detección es elevado debido a la baja eficiencia de medición.

En este trabajo, se propone la determinación simultánea de los niveles de ^{226}Ra y ^{210}Pb en aguas por centelleo líquido. Previamente los radionucleidos son separados en procesos que son rápidos, sencillos y que utilizan reactivos accesibles y de bajo costo. Se optimizó la separación y medición de ^{210}Pb y se incluyó la técnica de medición de ^{226}Ra ya usada por nuestro laboratorio en el proceso de determinación de ^{210}Pb. Asimismo, esta técnica permite también realizar la determinación de ^{210}Po.

MATERIALES Y MÉTODOS

La técnica propuesta para ^{210}Pb, es el resultado de la optimización tanto de los procesos químicos de separación como del método de medición de una técnica que utiliza contaje por centelleo líquido[7].

La técnica se basa en la precipitación del ^{210}Pb y ^{226}Ra como sulfatos. Posteriormente ambos radionucleidos se disuelven al formar complejos por el agregado de EDTA (ácido etilendiaminotetracético). Luego de acidificar el medio con HAc (ácido acético), los sulfatos de radio y bario vuelven a precipitar, mientras que el complejo del plomo es lo suficientemente estable como para persistir en estas condiciones. De esta manera son separados ambos radionucleidos. En la Figura 1 se detalla el esquema del proceso.

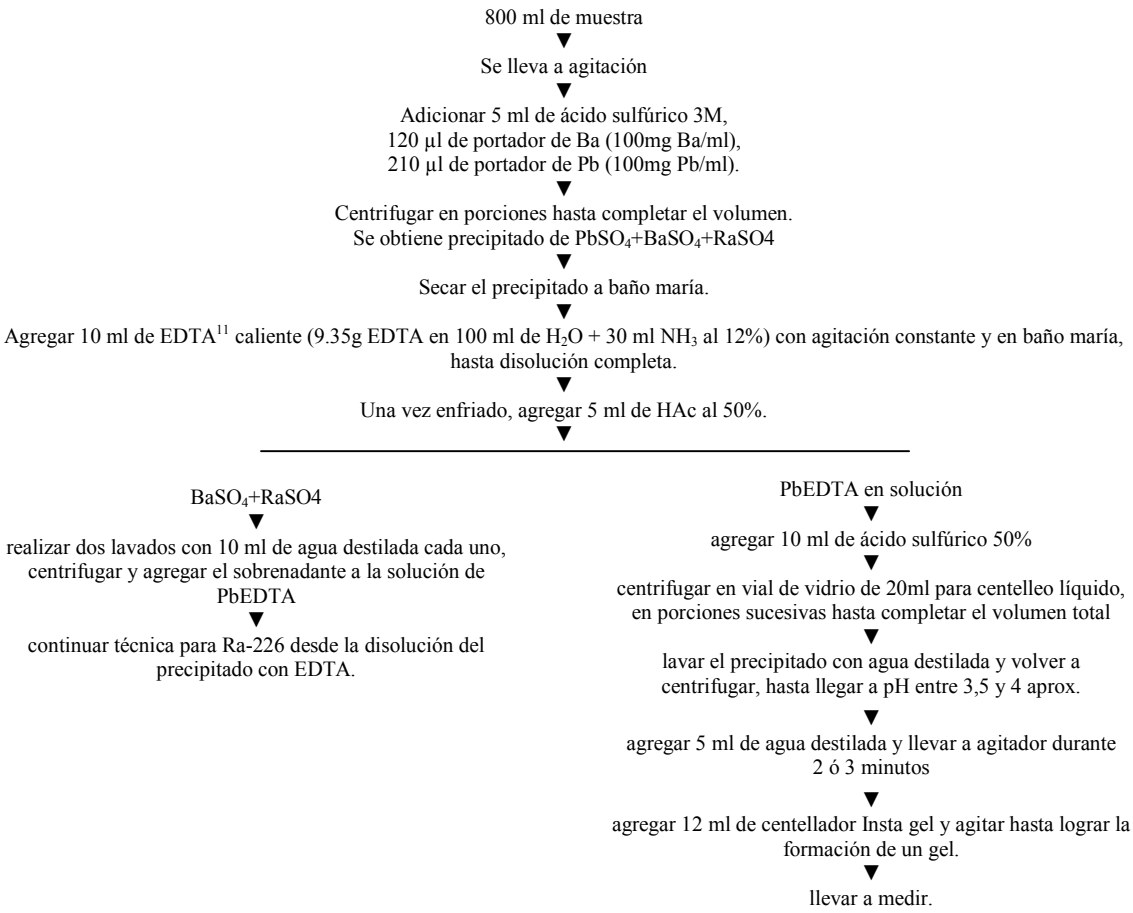

Figura 1

Las muestras a ser analizadas se deberán tomar en botellas de polietileno a las que previamente se agregará ácido nítrico concentrado para obtener soluciones 0.2M de ácido, para evitar, de esta manera, los efectos de absorción.
Para esta técnica se utilizaron reactivos de grado analítico. Los viales empleados para la medición son viales de vidrio para centelleo con bajo contenido de ^{40}K. La primer serie de centrifugaciones se realiza en recipientes de 250 ml a 1600 rpm, y la segunda serie de centrifugaciones en los mismos viales que luego se llevarán a medir, a 2000 rpm. El centellador utilizado es Instagel XF. La medición de ^{210}Pb se realizó en un equipo de centelleo líquido de bajo fondo, Packard modelo Tri-Carb 2250 LL.
Esta técnica permite también la determinación de ^{210}Po. La misma se realiza sobre el sobrenadante obtenido de la primer serie de centrifugaciones. Su aislamiento se logra por deposición espontánea del radionucleido sobre discos de plata y luego se realiza la medición por espectrometría alfa. Al momento de escribir el presente trabajo esta técnica se encuentra en etapa de optimización.

Para la preparación tanto de blancos como de muestras patrones se partió de 800 ml de agua destilada a la cual se le realizó todo el proceso descripto anteriormente. En el caso de las muestras patrón utilizadas para calcular el valor de eficiencia, se les agregó aproximadamente 1g de solución patrón a cada muestra. El patrón utilizado fue obtenido por dilución de una solución patrón de ^{210}Pb (Analitics) en HNO_3 0,1M.

RESULTADOS

A los fines de ajustar esta técnica, se determinaron en forma experimental los valores óptimos de las distintas variables que se detallan a continuación.

En el caso de los portadores, la cantidad agregada debe ser suficiente como para precipitar cuantitativamente el ^{210}Pb y el ^{226}Ra, pero no excesiva para luego poder ser disuelto con un volumen adecuado de EDTA. El agregado de ácido y portadores debe realizarse en el orden y condiciones indicadas en el esquema para asegurar la completa precipitación de los radionucleidos.

El rendimiento de la primer precipitación fue determinado por gravimetría utilizando distintos volúmenes de H_2SO_4 3M, encontrándose que para 5 ml el porcentaje de recuperación del plomo fue de 90 ± 4 %. Este fue el valor óptimo encontrado.

Respecto al EDTA, se ensayaron volúmenes de 5 y 10 ml, observándose que para 10 ml la disolución era completa, no así para 5 ml.

En cuanto al volumen de HAc necesario para la precipitación selectiva de ^{226}Ra, se midió la eficiencia total de ^{210}Pb obtenida utilizando 5, 10 y 15 ml, logrando valores de: 0,71 ± 0,07; 0,61 ± 0,05 y 0,66 ± 0,07 respectivamente. Con estos resultados se optó por utilizar un volumen de 5 ml.

Desde la etapa del agregado de EDTA, se minimizó el volumen total de solución, para disminuir la cantidad de centrifugaciones que se deben realizar luego de la segunda precipitación con H_2SO_4, debido a que se realizan en viales de 20 ml.

En cuanto a la precipitación final de $PbSO_4$ se ensayó con 5 y 10 ml de H_2SO_4 50%, obteniéndose resultados de eficiencia total de ^{210}Pb de: 0,65 ± 0,06 y 0,71 ± 0,07 respectivamente. En vista de estos resultados, y para asegurar la precipitación total del plomo, se eligió utilizar un volumen de 10 ml.

Es importante destacar que el crecimiento de cristales influye de forma negativa, ya que el gel final que se llevará a medir debe ser homogéneo, y esto se logra con un precipitado fino de sulfato de plomo.

Se optimizó la relación volumen de agua a volumen de centellador para obtener un gel estable.

Con la técnica optimizada, se calculó la eficiencia para ^{226}Ra a partir de los estándares preparados a tal fin. Se midió según la técnica ya utilizada por este laboratorio[6] verificándose una eficiencia de 3,5 ± 0,4 cpm/dpm.

La eficiencia determinada para el ^{210}Pb en la ventana optimizada (rendimiento químico más eficiencia de medición) fue de 0,54 ± 0,06 cpm/dpm (ver Fig. 2). La ventana se seleccionó en base al mayor valor del factor de mérito, y resultó ser de 6 a 20,5 keV.

Valor Ef. (cpm/dpm)	**frecuencia**	**Valor Ef. (cpm/dpm)**	**frecuencia**
0.43	1	0.55	3
0.44	1	0.56	3
0.46	2	0.57	2
0.47	1	0.59	2
0.53	1	0.60	1
0.54	2	0.62	3
Valor Ef. Promedio (cpm/dpm)		0.54	
Desviación estándar (cpm/dpm)		0.06	
Cantidad de datos		22	

Figura 2

El tiempo de medición utilizado es de 100 minutos, ya que proporciona un límite de detección adecuado. Las muestras se miden inmediatamente luego de agregar el centellador, para minimizar el crecimiento de ^{210}Bi. Al respecto se encontró que se puede despreciar este crecimiento en la ventana de medición dentro de las doce horas transcurridas desde la separación del plomo. De esta manera se pueden realizar series de varias muestras para medir dentro de este período de tiempo.

En la Figura 3 se observa un espectro obtenido luego de procesar una muestra contaminada con ^{210}Pb y ^{226}Ra, donde se puede ver el pico y la ventana seleccionada. También se observa que no se evidencian rastros de ^{226}Ra en los espectros finales, confirmando que hemos separado eficazmente ambos radionucleidos.

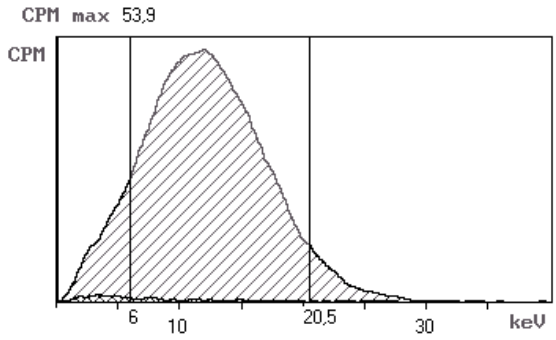

Figura 3

La mínima actividad detectable (*MAD)* se calculó en base a la siguiente fórmula[12]:

$$MAD\ (Bq/l) = \frac{3 + 4{,}65 S_0 T}{E_f V T 60}$$

donde:

S_0 desviación estándar de los blancos
T tiempo de medición
E_f eficiencia de medición de ^{210}Pb en la ventana elegida
V volumen de muestra utilizado

Se obtuvo un valor igual a 0,06 Bq/l, para un blanco promedio de 6,69 cpm con una desviación estándar de 0,32 cpm; una eficiencia promedio de 0,54 cpm/dpm; un volumen de 0,8 l y 100 minutos de medición.

CONCLUSIONES

Se ha optimizado una técnica para la separación y posterior determinación de los radionucleidos ^{226}Ra y ^{210}Pb en muestras de aguas. La técnica de separación es rápida, se puede completar todo el proceso y comenzar la medición de ^{210}Pb en seis horas. Debido a esto se puede medir el ^{210}Pb cuando aún no ha crecido el ^{210}Bi, evitándose así su interferencia. La centrifugación final se hace directamente en los viales en los cuales se llevará a cabo la medición, por lo cual se evitan trasvases de precipitados que pueden llevar a pérdidas de rendimiento. La separación del ^{226}Ra se hace en forma eficiente, y se puede acoplar perfectamente la técnica de medición utilizada habitualmente por el laboratorio para este radionucleido con esta técnica de separación. Todos los pasos involucrados en el proceso son sencillos y utilizan reactivos accesibles y económicos. Todos estos motivos hacen que la técnica sea conveniente para determinaciones rutinarias.

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DNA Repair Capacity as a Predictive Radiation Sensitivity Test Inferences for Clinical Practice and Radiation Protection

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DNA REPAIR CAPACITY AS A PREDICTIVE RADIATION SENSITIVITY TEST INFERENCES FOR CLINICAL PRACTICE AND RADIATION PROTECTION

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ABSTRACT

Individual radiosensitivity is an inherent characteristic, associated with an abnormally increased reaction to ionizing radiation (IR). Human population is not uniform in its radiation sensitivity. Radiosensitive sub-groups exist, with an increased incidence of both deterministic and stochastic effects. The identification of such sub-groups should be relevant for radiation therapy and for radiation protection purposes. Large part of the spectrum of normal tissue reaction may be due to differences in individual radiosensitivity. Their occurrence and severity are mainly influenced by genetic susceptibility to IR.
Deficiencies in DNA repair mechanisms would be involved. Consequently, the characterization of DNA repair capacity in lymphocytes through cytokinesis blocked micronucleus (MN) and comet assays could be suitable approaches to evaluate in vitro individual radiosensitivity.
The aim of this study is to assess the in vitro radiosensitivity in peripheral blood lymphocytes of 54 cancer patients prospectively, retrospectively and pre-radiotherapy studied, using MN and comet assays, in comparison with the observed clinical radiation reactions to predict adverse side effects.
The pre-radiotherapy ex vivo radiation response data suggest that MN yield and the assessment of DNA repair kinetics in peripheral blood lymphocytes may be a predictive tests for the detection of patients with a greater than average risk of developing radiation toxicity. The differences between average-reactors and over-reactors were significant. Comet assay showed a good predictive potential.

INTRODUCTION

In the last decades, the cancer patient population that reaches prolonged survival has increased as a result of the improvements in cancer therapy and health care. 62% of adult and 77% of pediatric cancer patients survive further than 5 years and thus could be considered as chronic patients [1]. In this situation, adverse side effects in normal tissues unavoidably included within the treatment field are of particular interest.
Clinical studies have shown that a large part of the spectrum of normal tissue reactions may be due to differences in individual radiosensitivity. Their occurrence and severity are mainly influenced by genetic susceptibility to ionizing radiation (IR) [2].
Deficiencies in DNA repair mechanisms would be involved on hypersensitivity to the carcinogenic risk and to the deterministic effects of IR, leading to an increased risk of radiation-induced cancer and to severe deterministic effects of radiation. Consequently, the characterization of DNA repair capacity in lymphocytes through cytokinesis blocked micronucleus (MN) and alkaline single-cell microgel electrophoresis (comet) assays could be suitable approaches to evaluate in vitro individual radiosensitivity.

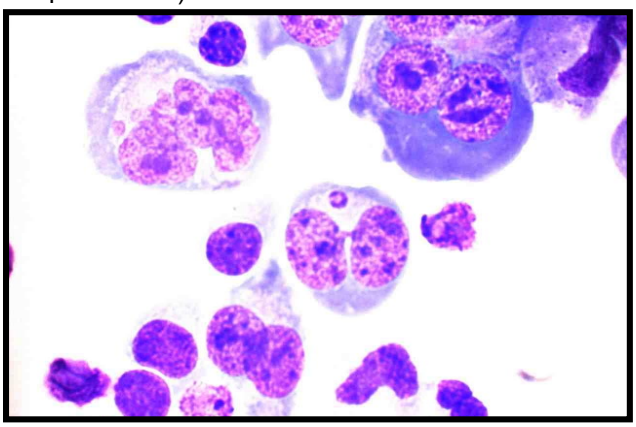
OBJECTIVE

The aim of this study is to assess the in vitro radiosensitivity in peripheral blood lymphocytes of cancer patients, using MN test and comet assay, in comparison with the observed clinical radiation reactions in order to predict adverse side effects

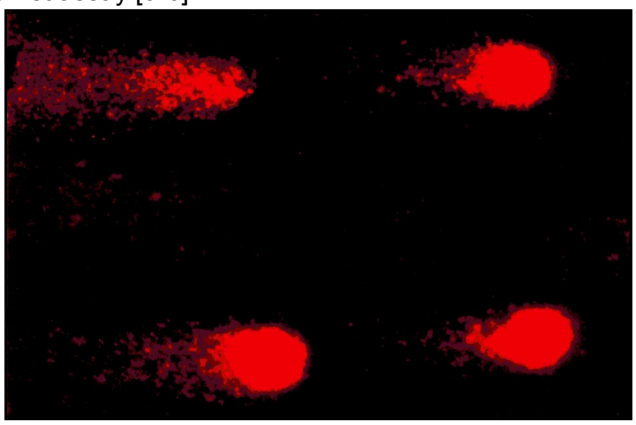
MATERIALS AND METHODS

- Blood samples from 54 cancer patients, with different tumor sites, receiving radiotherapy were evaluated prospectively, retrospectively and pre-radiotherapy.

- Prospective group: 19 patients were assessed by MN test [3]. Data were analyzed using a mathematical model [4] $F(MN)=\Sigma(MN_i \cdot e^{-di.k})$ to evaluate the attenuation of the cytogenetic effect, F(MN), as a function of the time, d_i, between a single exposure and blood sampling, estimating a cytogenetic recovery factor *k*.



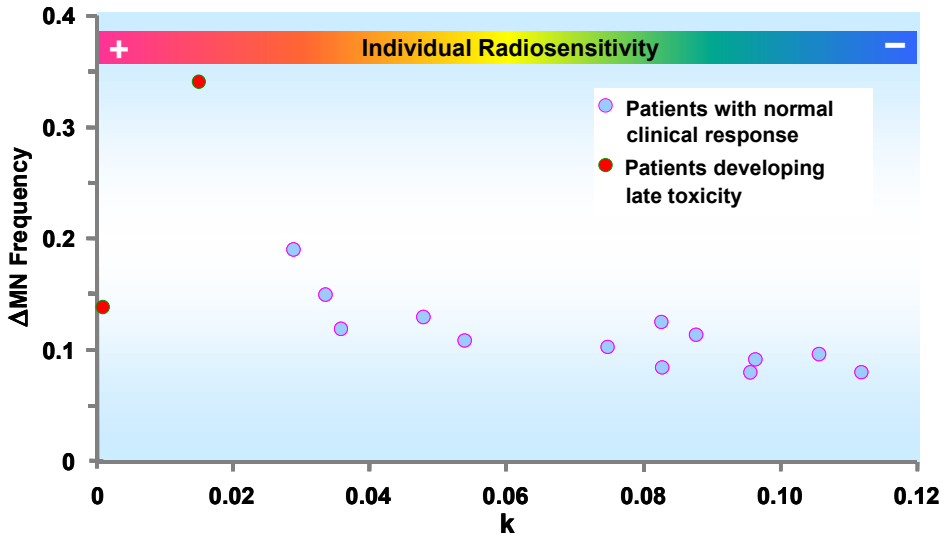
- Retrospective group: 19 patients were evaluated using MN test about 2-90 months after radiotherapy. Blood samples were irradiated in vitro with 2 Gy. MN frequencies from both, unirradiated and in vitro irradiated samples, were compared with frequencies derived from the calibration curve (healthy donors), using χ^2 test. One over-reactor and patients that did not develop late effects were also evaluated through comet assay [5-6].
- Pre-radiotherapy group: Samples from 16 patients were in vitro irradiated with 2 Gy. MN data were analyzed comparing expected MN frequencies with values observed after in vitro irradiation. DNA repair capacity was evaluated through comet assay for initial damage and after specific times of repair (5-120 minutes). Captured images were analyzed by CASP image analysis software [7] Repair capacity was quantified by the Olive tail moment [8], whose distribution was adjusted by a two-parameter Weibull model [9]. A non-linear regression analysis and curve fitting program (NLREG) was applied to asses the repair profiles.



RESULTS

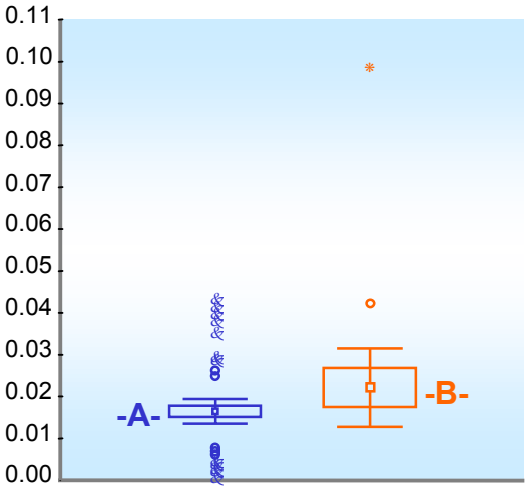
Prospective evaluation

ΔMN frequency vs cytogenetic recovery factor

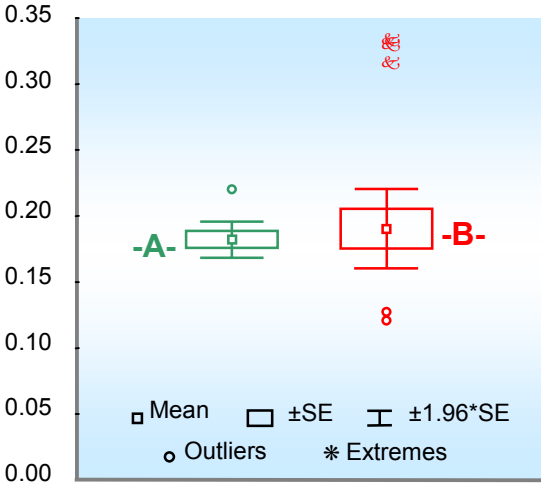


In 2 out of the 19 prospectively evaluated patients, the MN frequencies approach the calibration curve. This indicates low recovery from the cytogenetic effect and high radiosensitivity of the patients. Factor *k* correlates with the individual radiosensitivity. Patients with low recovery from the cytogenetic effect (*k* tending to zero) developed late toxicity (fibrosis and actinic rectitis).

Retrospective evaluation



Healthy individuals -A- and spontaneous cancer patients -B- MN frequencies from unirradiated cells

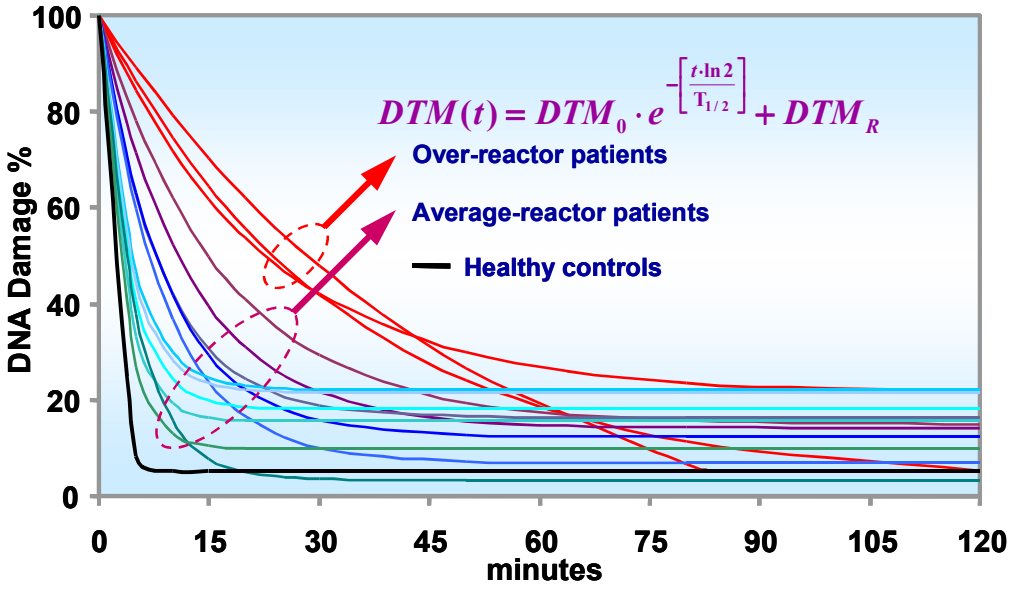


Healthy individuals -A- and cancer patients -B- MN frequencies after in vitro irradiation with 2 Gy

Lymphocytes of 3 out of 4 patients that had developed late reactions were significantly more radiosensitive than lymphocytes from the rest of the patients and normal donors. The individual cytogenetic response suggests a correlation with the maximum grade of late reaction (osteonecrosis, fibrosis and trismus).

Pre-treatment evaluation

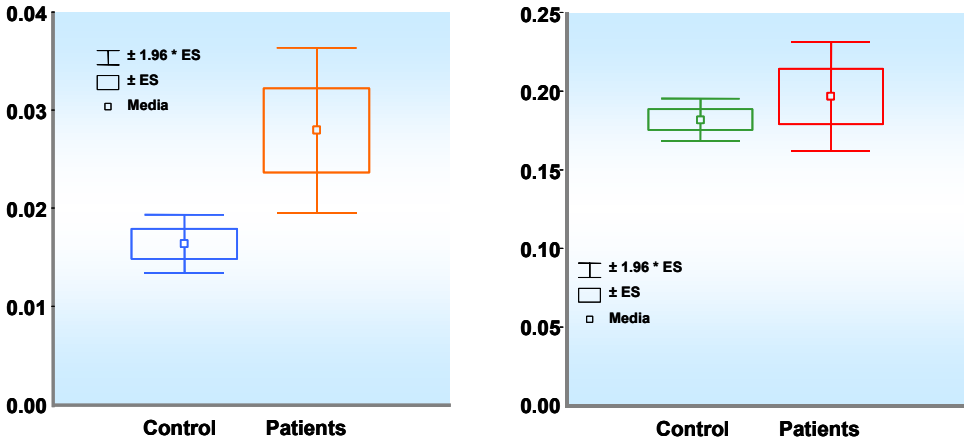
Repair kinetics in lymphocytes determined by the comet assay: Three subpopulations were identified, characterized by the means of their repair mean half-time: healthy controls (2.6 ± 0.3 minutes), average-reactor cancer patients (4.7 ± 2.9 minutes) and over-rector cancer patients (24.9 ± 10.4 minutes). Differences among means resulted significant, verified with Student's t-test.



DTM = Damage (Olive tail moment)
DTM_0 = Initial damage
DTM_R = Residual damage

$T_{1/2}$= Repair half time
t = Incubation time after in vitro irradiation

Spontaneous MN Frequencies MN Frequencies of irradiated cells



DISCUSSION

- In the prospective evaluation, the MN assay correlated with the clinical late toxicity. Nevertheless, the predictive potential of prospective MN assay resulted limited by the requirement to accumulate 2 Gy equivalent whole-body dose to find a substantial difference in the DNA repair capacity, measured through k parameter [10].
- In the retrospective evaluations, patients that had developed late tissue reactions showed both spontaneous and in vitro radiation-induced MN frequencies significantly increased compared with the expected values from the calibration curve. Applying comet assay, a significant difference between one over-reactor (osteonecrosis) and average-reactors was found.
- In the pre-radiotherapy evaluations, experimental data showed, for both MN and comet assays, significant differences among healthy controls, average reactor and over reactor cancer

patients, suggesting that the MN yield and the assessment of DNA repair kinetics in the lymphocytes of patients may be predictive tests for the detection of patients with a greater than average risk of developing radiation toxicity. Using comet assay we have identified three subpopulations characterized by their repair mean half-time: healthy controls (2.6 ± 0.3 minutes), average-reactor cancer patients (4.7 ± 2.9 minutes) and over-rector cancer patients (24.9 ± 10.4 minutes)

CONCLUSIONS

MN and comet assays correlated with the clinical radiation signs of radiation sensitivity and thus, they may play the role of a warning signal when deficiencies in repair capacity are detected.

The impact of the individual radiosensitivity would require special consideration for patients, occupationally exposed workers and victims of radiation accidents.

Therefore, should the presently available in vitro radiosensitivity tests be applied in cases of suspected hyper-radiosensitivity syndromes, family history of cancers, children with cancer and a family history of cancer, etc. to tailor patient's radiation therapy? Should hypersensitive persons and hypersensitive cancer patients and their eventual siblings not be exposed or minimally exposed to radiations?

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Experiencia regulatoria argentina en gammagrafía industrial en materia de radioprotección

Ermacora, M.G.

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EXPERIENCIA REGULATORIA ARGENTINA EN GAMMAGRAFÍA INDUSTRIAL EN MATERIA DE RADIOPROTECCIÓN

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RESUMEN

La gammagrafía industrial siempre fue responsable de los mayores índices de accidentes radiológicos en casi todo el mundo. Esto se debe principalmente a las altas actividades de las fuentes radiactivas utilizadas, las cuales son transportadas constantemente en los equipos que las contienen entre el depósito y las áreas de trabajo, y a presiones de carga de trabajo que pueden inducir a descuidos en el seguimiento de los procedimientos de operación, si no existe una cultura de la seguridad arraigada.

El propósito de este trabajo es presentar los aspectos principales de la reglamentación argentina en materia de radioprotección para controlar esta práctica y de esta manera contribuir a reducir los riesgos asociados. Asimismo, se describen algunos incidentes ocurridos en Argentina durante los últimos años, las causas que condujeron a dichos sucesos y su relación con el incumplimiento a la normativa vigente, sus consecuencias y las medidas adoptadas para remediarlas.

Por último, se destaca la importancia del rol que cumplen la capacitación y el entrenamiento en el fortalecimiento de la Cultura de Seguridad, elemento clave en todo emprendimiento.

ABSTRACT

Industrial gammagraphy has always been responsible for the highest rates of radiological incidents in almost every part of the world. This is mainly due to the high activities of the radioactive sources used, which are constantly transported in the equipment that contains them between the storage and the areas of work, and is also due to work load pressures which may induce to negligence in following the operational procedures, if a strong Safety Culture does not exist.

The purpose of this paper is to present the main aspects of the argentine regulations relating to radiation protection to control this practice and to contribute in this way to reduce the associated risks. In addition, some incidents occurred in Argentina during the latest years, the causes that led to those events and their relation to the inobservance of the regulations in force, their consequences and the measures taken to repair them are described.

Finally, the importance of the role that education and training has in the strengthening of Safety Culture, key element of all undertaking, is highlighted.

INTRODUCCIÓN

La gammagrafía industrial es una técnica de ensayo no destructivo, destinada al control de calidad de materiales, muy utilizada en las industrias siderúrgica, naval, nuclear, petrolera, etc. Es empleada principalmente con el objeto de identificar y clasificar defectos de soldaduras. Esta técnica es llevada a cabo con fuentes selladas de radiación gamma de Iridio-192, Cobalto-60, Cesio-137, Se-75, Iterbio-169 y Tulio-170, siendo el primer radionucleido mencionado el de uso más extendido en nuestro país. Estas fuentes son alojadas en unidades fijas, móviles o portátiles.

Las unidades fijas no generan mayores problemas en radioprotección. Por el contrario, la gammagrafía con unidades portátiles puede conducir a un nivel de exposición más importante y presenta riesgos potenciales de sobre-exposición. En nuestro país, este tipo de unidades son las más utilizadas en las prácticas de gammagrafía.

Un ejemplo de una fuente de Iridio-192 de 3,7 TBq da una tasa de dosis de 20μSv/min (1,2 mSv/h) a 20 m, que representa un moderado nivel de exposición para una persona que pasara por el perímetro de seguridad. La misma fuente da 8 mSv/min a 1 m, que es el nivel de dosis potencialmente recibida por los operadores en el caso de un rescate de fuente. Pero a 1 cm la tasa de dosis de 80 Sv/min produciría exposiciones severas.

Debemos recordar el accidente ocurrido en nuestro país en el año 1968, en el que un trabajador encontró una fuente de Cs-137 de 0,5 TBq y sin reconocerla como tal la llevó en el bolsillo de su pantalón durante 18 horas. El operador del equipo no advirtió que había perdido la fuente en una destilería de petróleo. Como consecuencia de este hecho, una persona resultó irradiada habiendo sufrido lesiones severas, úlceras en la mano derecha y la amputación de ambas piernas. Se estimó una dosis máxima en piel de 17.000 Gy, en gónadas de 20 Gy y en todo el cuerpo de 0,5 Gy. La falta de seguridad de la fuente condujo a estos resultados.

El objetivo de este trabajo es presentar los aspectos principales de la reglamentación argentina para controlar esta práctica y reducir los riesgos asociados, describir algunos incidentes ocurridos en Argentina durante los últimos años y destacar la importancia de la capacitación y el entrenamiento en la promoción de una fuerte cultura de la seguridad.

DESARROLLO

EL CONTROL Y LA FISCALIZACIÓN EN ARGENTINA

NORMATIVA VIGENTE

La Ley N° 24804, denominada Ley Nacional de la Actividad Nuclear, sancionada el 2 de abril de 1997, y su Decreto Reglamentario 1390/98, establece que el Estado Nacional fijará la política nuclear y ejercerá las funciones de regulación y fiscalización por medio de la Autoridad Regulatoria Nuclear (ARN). Entre otras funciones, esta Ley confiere a la ARN la competencia de dictar normas regulatorias sobre seguridad radiológica y transporte de material radiactivo; otorgar, suspender y revocar licencias y permisos; y realizar inspecciones y evaluaciones regulatorias. En el caso particular de gammagrafía industrial, los requisitos que deben ser cumplidos en materia de seguridad radiológica en este campo de aplicación están establecidos en las Normas AR 7.9.1 Rev.1 “Operación de equipos de gammagrafía industrial”, AR 7.11.1 Rev 1 “Permisos individuales para operadores de equipos de gammagrafía industrial”, AR 10.16.1 Rev.1 “Transporte de materiales radiactivos” y AR 10.1.1 Rev. 3 “Norma Básica de Seguridad Radiológica”.

A continuación se detallan los elementos principales sobre los que se centra la actividad regulatoria argentina relacionada con la gammagrafía industrial, y se describen algunos incidentes ocurridos relacionados con los mismos.

➔ EL EQUIPO

En general un equipo de gammagrafía industrial consta de un contenedor de alojamiento y transporte del material radiactivo, una fuente radiactiva, un telemando, y un tubo guía.

El proyector o contenedor debe estar correctamente identificado mediante dos o más placas metálicas con la siguiente información grabada o estampada en forma visible: marca, modelo y número de serie, radionucleido contenido, máxima actividad del radionucleido autorizado en el proyector, dirección del fabricante, símbolo normalizado de radiación y la palabra “RADIACTIVO”.

Otra placa, adosada al proyector o contenedor, debe mostrar el símbolo químico y número másico del radionucleido contenido, la actividad, fecha de calibración, nombre del fabricante, modelo y número de serie de la fuente.

El responsable de la Licencia debe contar con un certificado emitido por el fabricante del equipo que permita identificarlo, y con el manual de operación y mantenimiento, así como con un registro que incluya los controles realizados sobre cada equipo.

En muchos casos, es el mal estado de piezas mecánicas el que condujo a la pérdida de control de la fuente radiactiva con los consecuentes riesgos para una persona que pudiera tomar la fuente en su mano.

Existen escenarios que pueden presentarse y que pueden conducir la operación a este tipo de incidentes, algunos de ellos evitables y otros muy difíciles de evitar. Entre los primeros están:

- que se haga difícil el accionamiento de la manivela por falta de limpieza y lubricación inadecuada del cable flexible.
- que se debilite la sección del cable flexible debido al uso, con lo cual aumenta la probabilidad de un corte del mismo.
- que las conexiones entre cable y portafuente se encuentren desgastadas o dañadas con lo que existe la probabilidad de que una maniobra brusca produzca su desenganche.
- que la manguera se encuentre aplastada, con lo que se puede trabar el movimiento del cable flexible, tanto del telemando como del tubo guía. En este último caso la fuente radiactiva podría quedar atrapada sin poder ingresar a su blindaje.

Existen también los imponderables como puede ser la caída de un elemento pesado sobre la manguera que produzca su aplastamiento e impida el retorno de la fuente a su posición segura, pero este problema se encuentra acotado mientras se lleven a cabo los procedimientos de rescate adecuados.

Estos escenarios pueden provocar un incidente, pero su ocurrencia no necesariamente lleva a la ocurrencia de un accidente. Es la concomitancia de alguno de ellos con fallas en el procedimiento de monitoreo de las tasas de dosis en contacto con los equipos lo que puede producir una pérdida de control de la fuente sin el conocimiento de que ello haya ocurrido.

➔ LA FUENTE RADIACTIVA

La fuente radiactiva está formada por material radiactivo de una alta actividad y se encuentra contenida dentro de cápsulas de acero inoxidable selladas, calificada como material radiactivo en forma especial y como fuente sellada para el uso específico. La misma se encuentra dentro de un portafuente, cuyo extremo no activo sobresale en parte del contenedor, y en el mismo se encuentra grabado el número de serie de la fuente.

Es fundamental efectuar el seguimiento de las fuentes. Es por ello que la Norma AR 7.9.1 establece un plazo de cinco días para que el titular de licencia comunique toda compra, venta, alquiler, préstamo o baja de las mismas. Se dispone de una base de datos, la cual permite conocer en cada momento en qué empresa se halla determinada fuente. De esta manera, la fuente es seguida desde la entrega hasta la disposición final en el área de gestión de residuos o hasta que sean exportadas a su destino de origen en el caso de las fuentes que hubieran sido importadas. Asimismo, el titular de licencia debe mantener un registro de movimiento que permita identificar en qué equipo se encuentra cada fuente y su ubicación en el lugar de las prácticas.

Por otra parte, cada pérdida o robo de fuentes debe ser declarado a la ARN, según lo requerido en el punto 62. a. de la Norma AR 7.9.1.

En la Argentina hay actualmente un fabricante y un importador de fuentes de Sudáfrica.

➔ EL INSTRUMENTAL DE RADIOPROTECCIÓN

La Norma AR 7.9.1 establece que el titular de la Licencia debe asegurar que los operadores tengan a su disposición medidores de radiación portátiles cuantitativos que puedan medir tasas de dosis equivalente ambiental entre 0 y 100 mSv/h como mínimo, monitores portátiles con indicación acústica, cuya tasa de repetición de pulsos sea proporcional a la tasa de dosis, dosímetros individuales integradores de lectura directa (tipo lapicera o similar), y dosímetros individuales de lectura diferida.

Asimismo, el titular debe llevar un registro del inventario del instrumental en su haber, y de las calibraciones, controles y mantenimientos efectuados.

➔ EL TRANSPORTE

El transporte de equipos de gammagrafía con sus fuentes de radionucleidos gamma emisores está bajo la reglamentación general sobre el transporte de materiales radiactivos contemplado en la norma AR.10.16.1.

Los contenedores con el material radiactivo se denominan bultos. Los bultos de transporte utilizados en gammagrafía industrial son del tipo B(U) o A. El tipo de bulto requerido es dependiente de la actividad que deba transportar y en cualquiera de los dos casos, tipo B(U) o A, antes de su aprobación como tales, especímenes de los mismos deben ser sometidos a exigentes ensayos que permiten inferir que son adecuados para resistir fuerzas de impacto grave, fuerzas de aplastamiento, inmersión en líquido y tensión térmica sin pérdidas de los contenidos radiactivos ni pérdida significativa del blindaje. El titular de licencia debe poseer el certificado de aprobación por parte de la autoridad competente.

El transporte de estos contenedores cargados debe realizarse de acuerdo a las normas de transporte vigentes, convenientemente etiquetados de acuerdo a las tasas de dosis que producen, con la documentación correspondiente y en vehículos señalizados y adecuados para tal fin, con las trabas mecánicas necesarias para evitar su desplazamiento.

El transporte más crítico es aquél que es realizado por el operador, que viaja de un sitio a otro, en general en su propio vehículo. Esta clase de transporte da lugar a riesgos de pérdida (o extravío) o robo. En un primer momento es el vehículo el que es objeto de robo y luego, cuando los ladrones comprenden que hay una fuente radiactiva en el equipo, la abandonan en cualquier sitio. Otro riesgo son los accidentes de tránsito, y en este caso, aún si el equipo no es dañado, su presencia puede llevar a todo tipo de reacciones por parte de las personas que deben intervenir.

Cabe destacar dos episodios ocurridos en nuestro país en los años 2001 y 2003.

♦ El 9 de junio de 2001, en la ciudad de Neuquén, se produjo el extravío en la vía pública de un equipo de gammagrafía industrial conteniendo una fuente radiactiva de Ir-192 con una actividad de 1,96 TBq (53 Ci). El equipo, que era transportado en la parte trasera abierta de una camioneta, sujetado a la caja del vehículo con una soga, cae del mismo sin que lo advierta el conductor mientras se dirigía a la obra donde prestaría servicios.

Poco después lo recoge un albañil en la vía pública, quien lo confunde con un aparato vibrador para compactar hormigón. Dicha persona no comprende el idioma (inglés) de las identificaciones del equipo y al cabo de dos horas y media lo deja depositado en el patio trasero de la vivienda de su hija.

Por ese entonces, hacía media hora que la empresa de gammagrafía había notificado del hecho a la ARN, por lo que ésta le requirió a la misma, propietaria del equipo, que diera alerta a la población a través de los medios de difusión masivos. También se le requirió a la empresa que iniciara una búsqueda sistemática del equipo de gammagrafía utilizando el equipamiento de medición de radiaciones ionizantes que disponía.

La familia que tenía el equipo toma conocimiento, a través de la televisión, del aviso de alerta a la población. Poco después un integrante de la familia se comunica con personal de la empresa de gammagrafía para informarle que tenía en su poder el equipo extraviado. Finalmente, ocho horas después del extravío, la empresa recupera el equipo.

La ARN a través del Sistema de Intervención en Emergencias Radiológicas (SIER), llega al lugar –a 1000 km de Buenos Aires- aproximadamente una hora después de haber sido recuperado el equipo y verifica el estado seguro del mismo y que la fuente radiactiva se encuentra alojada en su interior.

Se reconstruyen los hechos y preventivamente se toman muestras de sangre para efectuar dosimetría citogenética de las tres personas del público involucradas, sin que se hayan observado efectos de las radiaciones sobre las mismas.

♦ El segundo caso ocurrió en la ciudad de Río Grande, Tierra del Fuego, el día 19 de noviembre de 2003. La ARN recibió un llamado telefónico de un operador de una empresa de gammagrafía industrial, que realizaba trabajos en un gasoducto del lugar, denunciando el extravío de un equipo de gammagrafía con una fuente de Ir-192 de actividad 0,24 TBq (6,5 Ci) a la fecha de ocurrido el hecho. El equipo era transportado en una camioneta alquilada, en la

caja abierta, y sobre un carro con ruedas. Personal del Sistema de Intervención en Emergencias Radiológicas (SIER) de la ARN se hizo presente en el lugar del hecho al día siguiente. Inmediatamente se realizó una conferencia de prensa para dar amplia difusión a todos los medios de comunicación locales. La ARN se abocó a la búsqueda del equipo, con la colaboración de la Policía de Tierra del Fuego, Gendarmería Nacional, Policía Federal, Defensa Civil, Prefectura Naval, el Juzgado Provincial y el Juzgado Federal. Todos los llamados recibidos denunciando sobre el equipo en cuestión fueron investigados sin excepción. En todo momento se mantuvo contacto con el Centro de Control de Emergencias de la ARN en Buenos Aires, para evacuar consultas técnicas y jurídicas, así como para solicitar apoyo adicional. Paralelamente, se realizó el asesoramiento de todos los servicios médicos de la ciudad, públicos y privados, dejando copias de dos instructivos: "Cómo reconocer y dar una rápida respuesta a una radiolesión accidental" (OIEA-OMS) y "Guía práctica para la rápida identificación de fuentes radiactivas y equipos que las contienen" (ARCAL-OIEA). Las tareas estratégicas específicas coordinadas con la Brigada de Investigaciones de la Policía Judicial dieron los resultados esperados y se recibió un llamado telefónico a las 03:30 horas del domingo 23 de noviembre en la Seccional de la Policía Judicial, diciendo que había sido visto el equipo en cuestión en el Barrio Austral en Río Grande. Junto con personal policial, el grupo de intervención se dirigió allí y realizó una verificación visual reconociendo el equipo en un contenedor de residuos. La zona se encontraba vallada por la Policía 20 metros alrededor del contenedor. El grupo de la ARN procedió a realizar las mediciones correspondientes y verificó que el punto de máxima exposición era de 60 μSv/h, valor compatible con la actividad de la fuente ubicada en su lugar de guarda dentro del equipo. Las mediciones de contaminación resultaron negativas. Se procedió a realizar una conferencia de prensa ante los medios de información periodísticos locales dando por finalizada la emergencia. La ARN realizó las actuaciones regulatorias del caso para deslindar las responsabilidades.

Estos hechos no condujeron a exposiciones peligrosas porque en todos estos casos la fuente permaneció confinada en el proyector/contenedor. Sin embargo, esta clase de incidentes puede tener serias consecuencias si la fuente fuera extraída del contenedor y tomada por una persona durante varias horas.

Una evaluación de estos eventos permite identificar como la causa principal la falta de acatamiento estricto de la reglamentación vigente en materia de transporte. Sin embargo, es importante destacar que el respeto de la normativa en relación con la inmediata comunicación a la ARN sobre los mencionados incidentes ha permitido evitar consecuencias mayores. La implementación de la difusión a través de los medios de comunicación masiva con el fin de alertar a la población ha jugado un rol muy importante.

➔ LOS TRABAJADORES

La reglamentación relacionada con los trabajadores tiene en cuenta la capacitación y entrenamiento, el monitoreo de exposición y la aptitud psicofísica.

- El certificado de aprobación del curso habilitante es otorgado actualmente por una institución que dicta el curso para operadores de equipos de gammagrafía industrial. La mesa evaluadora está integrada por una persona con conocimientos en radioprotección, personal con experiencia en la operación de estos equipos, un médico y personal de la ARN. El certificado se entrega luego de haber aprobado una evaluación escrita y oral en los siguientes temas: Conceptos generales sobre las radiaciones ionizantes, interacciones de la radiación con la materia, principios de detección de la radiación, fundamentos de dosimetría de las radiaciones, efectos biológicos de las radiaciones, criterios y normas de protección radiológica, fundamentos de cálculo de blindajes, evaluación de accidentes y cultura de la seguridad. El examen también abarca el conocimiento de los diferentes tipos de equipos, la reglamentación sobre el uso de fuentes y su transporte y los procedimientos a seguir en caso de emergencias.

Es relevante resaltar la suma importancia que reviste la capacitación y el entrenamiento de los operadores, ya que gran parte de los incidentes que pueden derivar en accidentes se deben a factores humanos.

Por ejemplo, en el año 2004, en nuestro país, durante una de las inspecciones de rutina al depósito de una instalación se detectó que uno de sus equipos registraba una medición de tasa de exposición injustificadamente elevada, descubriéndose luego que se debía a que la fuente

de Ir-192 se encontraba en posición invertida dentro del canal en S del proyector, es decir que no se encontraba adecuadamente blindada. El responsable manifiesta haber solicitado a uno de sus operadores el trasvase de la fuente a un contenedor, y no haber chequeado el correcto traspaso, habiendo incurrido en incumplimiento de lo establecido en la norma “Operación de equipos de gammagrafía industrial”. Esto evidencia asimismo la falta de entrenamiento por parte del operador que realizó la maniobra.

- El monitoreo de la exposición comprende el uso de dosímetros individuales integradores de lectura directa y diferida. Por otra parte, el titular de licencia debe notificar a cada operador de su correspondiente informe dosimétrico mensual y enviar a la ARN los informes anuales de dosimetría.

- Asimismo, es necesario asegurarse que el operador esté en buen estado de salud con una buena condición física y psicológica. La Norma AR 7.11.1. establece que para solicitar o renovar un permiso individual el operador debe presentar un certificado de aptitud psicofísica aprobado por el médico examinador. Es muy importante que las prácticas de gammagrafía se realicen en equipos de dos o tres personas por razones de seguridad. La Norma Argentina establece que como mínimo debe haber dos personas operando.

INSPECCIONES REGULATORIAS

Con el objetivo de verificar el cumplimiento de los requerimientos de la normativa vigente en materia de radioprotección la ARN lleva a cabo inspecciones. En el caso particular de gammagrafía industrial, se pueden clasificar las mismas en dos tipos:

- Inspecciones de depósito: se verifica el lugar de almacenamiento de los equipos y contenedores.
- Inspecciones de campo: se controla la práctica propiamente dicha donde se radiografían los tubos y cañerías.

A continuación se describen los principales aspectos verificados durante las inspecciones en los depósitos:

- Correcta señalización del depósito.
- Tasas de dosis en las inmediaciones del mismo.
- Tasas de dosis en la superficie exterior de los contenedores.
- Estado de conservación del contenedor verificando su identificación y existencia de la placa identificatoria de la fuente que se aloja en su interior.
- Accionamiento de la llave de cierre del contenedor.
- Estado de los telemandos, tubos guía y demás accesorios.
- Instrumental de radioprotección.
- Estado del libro de registros de inventario y movimiento de fuentes y equipos, y de los registros dosimétricos del personal.
- Procedimientos de operación, de actuación en emergencias y de transporte.

En las inspecciones de campo se efectúan algunos de los controles mencionados anteriormente y además se realiza:

- Verificación del cumplimiento de las normas de transporte de material radiactivo.
- Verificación de que intervenga un mínimo de dos personas en la operación (un operador y un ayudante), y constatación de que al menos una de ellas posea permiso individual vigente para esta práctica, otorgado por la ARN.
- Verificación del uso del instrumental de protección radiológica tanto del operador como de su ayudante.
- Verificación del cumplimiento con los procedimientos de operación.
- Verificación de la señalización de la zona de trabajo.
- Monitoreo de los vallados.

El riesgo mayor en la operación normal en este tipo de prácticas es la irradiación externa de miembros del público o del personal ocupacionalmente expuesto.

Con el objeto de evitar los riesgos, que están relacionados con los altos niveles de radiación cuando las fuentes son proyectadas, existen procedimientos durante la operación que consisten en la demarcación y vallado de la zona de trabajo. El sistema consiste en alertar a los miembros del público sobre el riesgo radiológico a que se someterían de acercarse, tratando asimismo de impedirlo. La señalización se realiza con carteles que indican el uso de material radiactivo y el vallado mediante soga o cinta. No obstante el operador deberá ejercer un control visual de manera de verificar que no haya ninguna persona no autorizada dentro de la zona controlada.

Con respecto a los dos últimos puntos mencionados, cabe aclarar que la normativa regulatoria argentina no establece un límite de tasa de dosis para la delimitación de áreas de vallado, sino que deja a criterio del operador el establecimiento de dicho límite para cada caso particular, teniendo en cuenta para los cálculos de distancias el factor de ocupación, que suele ser muy bajo en los sitios donde las prácticas son llevadas a cabo. Por otro lado, con la utilización de elementos blindantes, como colimadores, estas distancias se reducen considerablemente permitiendo asimismo disminuir las dosis recibidas por el personal ocupacionalmente expuesto.

La frecuencia recomendable de la inspección, teniendo en cuenta que los equipos poseen fuentes radiactivas de considerable actividad y que en su mayoría son portátiles, es anual.

Durante el año 2004 se han totalizado 72 inspecciones, habiéndose realizado 12 de ellas en campo y las restantes en los depósitos de las 63 empresas relacionadas con la operación de equipos o fabricación, importación, exportación y venta de fuentes radiactivas selladas para gammagrafía industrial habilitadas en ese período en nuestro país.

PUNTOS A DESTACAR

Todo el esfuerzo regulatorio descripto ha permitido evitar desde 1969 hasta el presente accidentes de gran envergadura en Argentina.

Sin embargo, es relevante destacar los siguientes puntos, que contribuyen a una mejora continua promoviendo el arraigo de la Cultura de la Seguridad.

- Promoción de la comunicación de situaciones accidentales ocurridas por parte de los usuarios. Esto ha permitido intervenir y conducir las situaciones de emergencia eficientemente en el caso de involucrar áreas públicas.

- Difusión de información radiológica, incluyendo lecciones aprendidas, en el ámbito de las actividades industriales, promoviendo de esta manera la integración de la radioprotección con la seguridad e higiene laboral en general. El entrenamiento permanente es esencial al mantenimiento de un alto nivel de seguridad, lo cual requiere facilitar a trabajadores y otros representantes de partes involucradas el acceso a cursos y talleres. Al respecto, la ARN realiza reuniones informativas dirigidas a personal supervisor de industrias contratistas de empresas de gammagrafía industrial.

- La capacitación de los operadores de gammagrafía se realiza mediante la aprobación de un curso cuyo período de validez es de 2 años. Sin embargo, es importante reforzar el entrenamiento de los trabajadores en sus compañías. Se está evaluando solicitar a cada Entidad con Licencia la implementación de registros de entrenamiento del personal a su cargo (tanto operadores como ayudantes), en los que se incluyan copias de exámenes escritos, fechas y resultados de exámenes orales y prácticos y de las evaluaciones periódicas del trabajo, y nombres de las personas que condujeron los mismos. El entrenamiento podría tener una frecuencia anual como mínimo y es recomendable que el Responsable evalúe el trabajo de los operadores cada seis meses.

- Por otra parte, es esencial destacar la capacitación y el re-entramiento de los reguladores.

CONCLUSIÓN

El análisis de las situaciones que han terminado en incidentes o accidentes demuestran que los mismos se producen principalmente por ignorancia de los efectos de la radiación, falta de mantenimiento preventivo, no seguimiento de los procedimientos de operación o falta de seguridad en el transporte. Todo esto está contemplado en la normativa argentina.

En conclusión, el respeto de la normativa vigente en nuestro país ha permitido limitar el número de accidentes relacionados con la práctica de gammagrafía industrial.

Se sugiere, sin embargo, prestar especial atención a los puntos destacados relacionados con la difusión de información, la capacitación continua y el re-entrenamiento, de modo de contribuir a una mayor conciencia de una Cultura de Seguridad.

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Manual on the Assistance of Persons Accidentally Exposed to Radiation

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MANUAL ON THE ASSISTANCE OF PERSONS ACCIDENTALLY EXPOSED TO RADIATION

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ABSTRACT

This paper describes a manual that was prepared within the scope of an ARCAL program and its instrumental role for the preparation of health personnel in Latin America and the Caribbean, as well as in other regions, in recognising and treating medical conditions caused by accidental exposure to ionising radiation. Besides a detailed description of radiation injuries, the importance of the manual relies also on the fact that it has a clinical approach and is written in Spanish, so filling a gap on the medical literature for the aforementioned region.

Keywords: manual, radiation exposure, Latin America and the Caribbean

1 INTRODUCTION

The increasing use of radiation sources in different fields in Latin America and the Caribbean makes it mandatory to consider the importance of an adequate and timely delivered response to radiation emergencies. Nowadays, an additional threat extensively considered by nuclear regulatory bodies and other authorities is the malevolent use of radio-nuclear agents. This also emphasizes the need for an appropriate response in order to minimize the impact of a radiation accident or incident.
On the other hand, in the aforementioned region there are countries where either the structural capacity for the response to radiation emergencies is poorly established or that do not have adequately trained human resources for that or both. Also, the regional health personnel in general have very limited information on the medical manifestations of radiation injuries. Besides, virtually no Spanish or Portuguese written medical didactic material with a clinical approach is available for the regional health personnel.
From February 1999 to December 2000 representatives of Argentina, Brazil, Chile and Cuba conducted the Regional Co-Operation Agreement for the Promotion of the Nuclear Science and Technology in Latin America and the Caribbean (ARCAL) Project RLA/9/031, "Medical Treatment in Cases of Radiation Accidents".
Besides making arrangements for a Radiopathology iterative regional cooperation system to favor the development of consensual strategies for the diagnosis and treatment of radiation injuries, the project produced the “Manual on the Assistance of Persons Accidentally Exposed to Radiation”. Next, we describe the structure of the Manual and comments on specific parts of its content are made.

2 STRUCTURE OF THE MANUAL

The goals of the Manual are to establish uniform criteria for the evaluation and treatment of persons accidentally exposed to ionizing radiation, to serve as a guideline for health professionals in charge of the diagnosis, evaluation and treatment of radiation injuries and to facilitate the specialized medical assistance of persons accidentally exposed to ionizing radiation.
The Manual’s structure comprises introductory chapters on its applicability, on the definitions of radiological emergencies and nuclear accidents, on basic information on units and

measurements, on occupational doses limits and on concepts of accidental exposure to ionizing radiation. We summarize below the medical subjects of the Manual.

2.1 The acute radiation syndrome

A correlation between whole-body range of doses and the severity and the clinical and laboratorial manifestations of the acute radiation syndrome (ARS) is presented, so that a quick dose estimation and medical prognosis can be established by the medical personnel assisting a possibly exposed individual.
From the clinical point of view, it is interesting to observe the orientation that is given for the anamnesis, the medical and laboratorial evaluations of ARS patients. Considerations on treatment goes into details such as doses of antibiotics for prevention and treatment of infections in neutropenic patients, the timely use of bone marrow growth factors, indications for procedures like bone marrow transplantation and the prescription of antihelmintic drugs, as parasitic infestation could be a peculiar problem in Latin America and the Caribbean.

2.2 Local radiation injuries

This section deals with the definition of local radiation injuries, its etiology with emphasis on accidental exposure to industrial radiography sources. Local radiation injuries are classified according to their clinical evolution. In order to allow dose reconstruction and determine the extent to which tissues and vital organs have been damaged, the diagnostic approach includes clinical evaluation along with biological and biophysical parameters. According to the severivity of the damage, different strategies are proposed for medical and surgical treatment, including topical and systemic pharmacological approaches, debridement of devitalized tissues, temporary and definitive covering. Psychological assistance and rehabilitation are also considered. .

2.3 The combined injury syndrome and associated conditions

In some radiation accidents there can be the concomitance of conventional injuries, such as thermal or chemical burns, trauma etc. In such instances, the term "Combined Injury Syndrome" (CIS) is used. Experimental tests in animals and also observations in some accidents, have shown that the association of conventional and radiation injuries (ARS, local lesions) significantly worsens the prognosis, so that this subject was also considered in the Manual.
The combined injury has had its importance stressed nowadays in virtue of the awareness of international authorities on the possibility of the malevolent use of radiation sources, as the so-called “dirty bomb”.
Associated conditions (irradiation and contamination, for instance) are also addressed in the Manual.

2.4 Triage

In radiation accidents two situations should be differentiated:
a) There are a limited number of victims and adequate medical assistance can be immediately offered to all persons;
b) Many persons are involved (workers or individuals of the public), demanding the use of catastrophe handling principles. For example: the accident at the Chernobyl Nuclear Central (1986) and in the city of Goiânia, Brazil (1987).
When there are many victims, a classification of the exposed persons is necessary in order to prioritising the assistance. That procedure is called "triage" and one can find in the Manual the clinical, dosimetric and biochemical variables for sorting patients and allocate resources according to the medical severity.

2.5 Radioactive contamination

The diagnosis, evaluation and treatment of internal and external contamination with radionuclides are discussed in this section of the Manual. The different physiopathologic phases of internal contamination are considered and very specific information (presentation, indication, posology, side effects, etc.) on an extensive list of different drugs is available. The same applies to other decontamination procedures.

The different behaviour of transportable and non-transportable elements is also discussed. Radiation protection procedures for personnel and equipments as well as the premises against radioactive contamination are available.

2.6 Planning and organisation of the response to radiation accidents

A radiation accident is a non-intentional event that may cause an overexposure, that is, irradiation or contamination with doses or incorporations greater than the legal permissible limits. Not every overexposure will cause a clinically evident injury. Radiological emergencies need a prompt response to minimize exposures to persons, to mitigate the consequences of the accident and to return conditions into normality. In such situations, the response capacity is not only a function of the available physical and human resources but also is dependent on a previous and adequate planning to guarantee efficacy and speediness. The handling of overexposed patients demands a multidisciplinary approach, involving different specialties in the Medicine, Radiopathology and Radiation Protection. Such aspects are discussed in the Manual.

2.7 Annexes

Ten annexes are complimentary to the Manual, as follows:

a) protocol for biological samples and early indicators (adults);
b) prevention and treatment of infections and hematological complications of the ARS;
c) drugs used for the treatment of internal contamination with radionuclides (generic name, trade name, how supplied, mechanism of action, dosage and administration, contraindication and adverse effects: 22 drugs);
d) preparation of decontamination solutions: 11 solutions;
e) list of minimally necessary equipment and material for the handling of contaminated persons;
f) criteria for medical intervention;
g) priorities for the medical assistance;
h) registration forms (5);
i) form for the notification of radiological accident;
j) registry of radiation accidents in Latin America and the Caribbean (local, year, source, dose/incorporation, number of significant exposures, kind of exposures and number of deaths).

3 COMMENTS AND CONCLUSIONS

Accidental radiation exposures pose a challenge concerning the moment of medical decisions aimed at modifying the evolution of the radiation induced pathology, as well as in respect to its prospective handling. Presently, although there are only seven commercial nuclear power plants under operation in Latin America, the 23 research reactors in activity in the region and the extensive use of radiation in the industry, medicine, research and other economical fields brings together a potential risk for incidents and accidents during the production, transportation or operations with radioactive material.

On the other hand, the medical preparedness and expertise to respond to radiation accidents in Latin America and the Caribbean is limited.

Retrospectively, we have found information on at least 23 important accidents in the region from 1962 to 2002, with 272 “significant” exposures (in accordance to the IAEA criteria [1]) leading to 32 deaths directly caused by radiation exposure. Moreover, it is likely that other non-notified or identified accidents have taken place in the region.

Besides, the most severe radiation accident in the occidental hemisphere happened in Goiânia [2], Brazil in 1987, with very important medical, psychological, environmental and socio-economical burdens.

Other significant accidents in the region which are worth mentioning are those that have occurred in El Salvador [3], Costa Rica [4], Peru [5], Panama [6] and Bolivia [7].

The “Manual on the Assistance of Persons Accidentally Exposed to Radiation” represents an endeavor for training health personnel in Latin America and the Caribbean (and in other regions too) in a wider context that should include measures in order to full capacitating health

professionals for the diagnosis and treatment of persons accidentally exposed to ionizing radiation.
Although not formally published as yet, this Manual has shown itself to be a very useful tool in training programs that have been developed in countries like Argentina, Brazil, Chile and Cuba.

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Medical Response to Radiation Emergencies in Argentina

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MEDICAL RESPONSE TO RADIATION EMERGENCIES IN ARGENTINA

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Abstract

Although radiation accidents are not frequent, the increasing use of radioisotopes in medicine and industry increases the likelihood of such accidental situations. Additionally, risks posed by the malevolent use of radiation sources have been highlighted during the last few years. In this context, the enhancement of national capabilities for medical assistance of victims in radiation emergencies becomes relevant. This communication describes the organization of medical response to radiation emergencies existing in Argentina.

A three-level system for medical response has been developed: pre-hospital response given on-site by local emergency services, assistance provided by emergency departments of local general hospitals and central reference hospitals for treatment of acute radiation syndrome, cutaneous radiation syndrome and internal contamination.

An education and training program is regularly executed at the three levels, including theoretical background as well as practical training. Guidelines and protocols for medical handling of victims have been elaborated and implemented. Research and development of new strategies for diagnosis and treatment of radiation injuries are promoted by ARN in close collaboration with physicians belonging to reference hospitals.

Keywords: radiation accidents, medical response

1 Introduction

The risks associated with radioactive sources have been the subject of increasing attention during the last decade. Accidental events that expose people to ionizing radiation are not frequent. However, the world's rising use of radioisotopes in medicine, research and industry and the growing reliance on nuclear power increase the likelihood of accidental situations. Moreover, the risks posed by the use of radioactive sources for malevolent purposes have been highlighted during the last few years. Experience has demonstrated a need for planning medical response for both pre-hospital and hospital management of victims of nuclear and radiological accidents [1]. This communication presents the organization of medical response in nuclear and radiological emergencies developed in Argentina.

2 General overview

Medical response for radiation accidents in Argentina has been organized in compliance with the federal legal framework. According to the Nuclear Federal Law N° 24804, the Nuclear Regulatory Authority/ARN is empowered to regulate and control the nuclear activity with regard to radiological and nuclear safety, physical protection and nuclear non- proliferation issues. The ARN has a system for intervention in radiological or nuclear emergencies with a primary intervention group, which is on duty in weekly shifts all year round. As part of this intervention system the ARN Radiopathology laboratory is in charge of:

- coordinating medical response in radiation emergencies;
- improving professional expertise to advise on diagnosis, prognosis and treatment of radiation injuries;

- performing research programs on radiopathology;
- making arrangements for providing specific supplies and equipments;
- elaborating guidelines and protocols for medical response; and
- implementing education and training programs for personnel involved in medical assistance of radiation casualties

In the event of radiation emergencies, technical support and advice concerning radiation monitoring, physical dose reconstruction, cytogenetic dosimetry and evaluation of internal contamination is provided by other ARN laboratories.

A three-level system for medical response has been organised:

1) pre-hospital response given on-site by local emergency services;

2) assistance provided by local general hospitals; and

3) healthcare at central reference hospitals for treatment of acute radiation syndrome, cutaneous radiation syndrome and internal contamination.

3 Pre-hospital response

Depending on the scenario of the accident, pre-hospital response may be given by the medical services of nuclear installations or by local emergency medical systems. While in many countries prehospital care systems are paramedical-based, prehospital response in Argentina includes physicians as part of the ambulance team. At this first-level, a conventional triage is performed with assessment and treatment of life-threatening injuries, followed by a radiological triage, implementation of initial decontamination and transportation of victims to the hospital emergency department, if necessary. As a result of the technical cooperation between ARN and the System for Emergency Medical Assistance/SAME, training activities are regularly organized and guidelines for on-site management and transportation of radiation victims have been included in the SAME procedures [2].

4 Medical response at local general hospitals

Emergency departments from local general hospitals are in charge of providing second-level response, including treatment of conventional and/or radiocombined injuries, continuation of external decontamination as necessary and treatment of prodromal symptoms. In the case of nuclear facilities, intermediate complexity-hospitals close to these installations have been selected with this purpose. A reception area for radiation victims is set up, taking into account radiation protection measures to avoid radioactive contamination spread [3]. The initial triage is completed at this level by a biomedical approach including physical examination, timing and severity of prodromal signs and symptoms, sequential blood counts and serum enzymatic levels (e.g.: amylase, glutamic oxalacetic transaminases/GOT, lactic dehydrogenase/LDH, alkaline phosphatase). Other biological samples may be collected at this level (e.g.: blood samples for cytogenetic analysis and HLA typing, nasopharyngeal swabs, urine, faeces). As a result of these evaluations, decisions are taken concerning the hospitalisation of victims or their transfer to a high complexity third-level hospital.

5 Medical response at central reference hospitals

Third-level response is offered by central high-complexity hospitals with suitable infrastructure, equipment, human resources and professional expertise for providing efficient healthcare to radiation victims. Radiopathology committees have been designed in these hospitals and protocols for diagnosis and treatment of acute radiation syndrome, cutaneous radiation syndrome and internal radioactive contamination have been elaborated [4,5]. Cooperation agreements have been signed between Nuclear Regulatory Authority and the following third-level hospitals: Navy Hospital, Burns Hospital and Buenos Aires University Hospital. These

agreements include scientific and technical cooperation for medical assistance of radiation victims, training programs and research activities on radiopatology.

The ARN promotes the interaction between the Radiopathology Laboratory and other national institutions with professional expertise concerning particular areas such as:

- Hematology: Bone Marrow Transplantation Units from 4 hospitals in Buenos Aires
- Toxicology: Argentine Toxicology Network/REDARTOX
- Pharmacology: School of Pharmacy and Biochemistry of Buenos Aires University
- Pediatry: Radiotherapy Department of the National Pediatrics Hospital
- Psychological impact in emergency and disaster situations: Human Factors Group of the System for Emergency Medical Assistance/SAME

6 Education an training

Education and training programs are regularly executed for physicians and nurses belonging to the national system of medical response in radiation emergencies, at the three levels. In collaboration with other institutions, ARN organizes courses including both theoretical and practical training on Medical Response in Radiation Accidents. People potentially involved in medical response in radiation emergencies participates in the exercises and drills organized by the ARN and nuclear facilities. Modules concerning radiation protection, radiation biology and radiopathology have been included in the syllabus of postgraduate courses for burns, toxicology and emergency specialists. Other kind of activities such as conferences, symposia and workshops are also promoted by the ARN to contribute toward the enhancement of quality of human resources in this area.

7 Regional Cooperation

The project RLA/9/031"Medical Treatment in Cases of Radiation Accidents" was conducted through February 1999 to December 2000 by representatives of Brazil, Chile, Cuba and Argentina in the framework of the Regional Cooperation Agreement for the Promotion of the Nuclear Science and Technology in Latin America and the Caribbean/ARCAL [6]. As a result of this project, a regional consensus approach concerning diagnostic and therapeutic strategies for radiation injuries was established. The first manual on Assistance of Persons Accidentally Exposed to Radiation written in Spanish was produced, with the double objective of providing guidance for medical management of radiation victims and having an useful tool for education and training in the region. A regional training course on “Medical Response in Radiation Accidents” was held in Buenos Aires in October 2000, with the participation of 14 physicians of the four countries.

The ARCAL project RLA/.9/045 "Enhancement and Harmonization of National Capabilities in Radiation Emergencies" was conducted through 2001 to 2004 by representatives of Brazil, Chile, Cuba, Venezuela, Mexico, Uruguay, Peru, Ecuador and Argentina. The workshop “Training the trainers on Medical Response in Radiation Accidents” was held in Buenos Aires in October 2003 in the framework of this project, with the participation of 19 professionals from Latin America and the Caribbean.

8 International Cooperation

As a liason center for the Radiation Emergency Medical Preparedness and Assistance Network (REMPAN) coordinated by WHO's Radiation Programme, the ARN Radiopathology Laboratory takes part of the international cooperation in nuclear or radiological emergencies. Within the international legal framework, this cooperation is addressed under the Joint Radiation Emergency Management Plan of the International Organizations [7], the Convention on Early Notification of a Nuclear Accident and the Convention on Assistance in the case of a Nuclear Accident or Radiological Emergency.

The ARN promotes the interaction between the Radiopathology laboratory and other international institutions for research cooperation on specific topics concerning radiopathology (Institut de Radioprotection et Sûreté Nucléaire/IRSN, Institut Curie and Hôpital Saint-Louis/CEA).

9 Final considerations

Medical planning is essential to cope with radiation accidents. This communication has described the organization of medical response in nuclear and radiological emergencies developed by the Nuclear Regulatory Authority/ARN in Argentina. As shown, medical response has been set up as a three-level system, taking account of the national needs and capabilities. Instead of having medical facilities for the sole purpose of treating radiation injuries, this system has been based on the adequation of pre-existing healthcare infrastructure, with emphasis on education and training of personnel potentially involved in medical care of radiation casualties. In this context, the ARN also promotes regional and international cooperation in order to enhance medical preparedeness and management in radiation emergencies.

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Argentine Intercomparison Programme for Personal Dosimetry

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ARGENTINE INTERCOMPARISON PROGRAMME FOR PERSONAL DOSIMETRY

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ABSTRACT

An Intercomparison Programme is being carried out in Argentina for individual monitoring services. The programme was designed to test, on a voluntary basis, the performance of the laboratories that provide individual monitoring services for X and gamma radiation fields in the range from low-level dose up to 100mSv. Irradiations were performed in full accordance with ISO 4037–3 recommendations by the Regional Reference Center for Dosimetry (CRRD), belonging to Atomic Energy Commission (CNEA) and the Physical Dosimetry Laboratory of the Nuclear Regulatory Authority (ARN). At the same time several items have been asked to each participant referring to the action range, the detectors characteristics, the laboratory procedures, the existence of an algorithm and its use for the dosimeter evaluation.
In this work the laboratories evolution performance throughout the program, based on ISO 14146 acceptance criteria are shown.

INTRODUCTION

Since 1997, an Intercomparison Programme for individual monitoring ~~it~~ is being carried out in Argentina. The programme is aimed at checking the performance of the laboratories for personal dose equivalent determinations, as well as to obtain information regarding the type of personnel dosemeter employed, the calibration procedures, and the proper use of radiological quantities. A similar program was performed in the past (1989 and 1990) by the former regulatory branch of the Atomic Energy Commission (CNEA) (1).

The programme was designed to test, on a voluntary basis, the performance of the laboratories that provide individual monitoring services for X and gamma radiation fields in the range from low-level dose up to 100mSv. It started with a meeting to introduce the quantities: Personal Dose Equivalent. For every exercise, each laboratory sent 15 personal dosemeters of the type issued in routine monitoring. Of these, 12 dosemeters were irradiated and the remaining 3 to be used as controls..

In Argentine, eighteen laboratories have been working on routine radiation surveillance. Nine of these are private and the rest of them belong to state laboratories. It is important to stress that beside the fact it is not a required attendance sixteen laboratories have participated.

DESCRIPTION OF THE PROGRAMME

Irradiation laboratories

Both the Regional Reference Center for Dosimetry (CRRD) ~~)~~ from CNEA and the Physical Dosimetry Laboratory of ARN performed the irradiation. The ^{137}Cs source of ARN has been calibrated by the CRRD in 1996, 2002 and 2004. The estimated total uncertainties associated with kerma free in air are of the order of 3.5% and 4% for CRRD and ARN respectively..

In 1996 a control of the CRRD was performed, with the TLD-Pill Box system by Physikalish Technische Bundesanstal (PTB), and by ARN. The difference between them was 4%. In view of these results ARN was designated as a verification laboratory by IAEA, for the irradiation of Secondary Standards Dosimetry Laboratories in the International Atomic Energy

Agency (IAEA) Latin America Intercomparison (1996-1998) (2), in IAEA RLA/9/041 Project (3) and played the same role in this national program.

Irradiation conditions

Irradiations were performed in full accordance with ISO 4037-3 (4) recommendations on ICRU phantom at normal incidence. Only in one exercise the angular response was tested for 0°, 30° and 60°. In order to investigate the energy response, the irradiations were made with X-ray ISO quality: W60, W80, W110 and W200, ^{137}Cs and ^{60}Co gamma rays and mixed fields of gamma and X-Ray and different qualities of X-ray.

Irradiations to evaluate the minimum detectable dose and the capability for high dose detection in case of an accident were made in the range from 0.3 mSv up to 100 mSv.

RESULTS

The ISO 14146 (5) about performance of personnel dosimetry laboratories was used and the results were analysed with the trumpet curved described by the following expression:

$$\left(\frac{1}{1.5}\right)\left[1-\frac{2Ho}{Ho+Hr}\right] \leq \frac{Hm}{Hr} \leq 1.5\left[1+\frac{Ho}{2Ho+Hr}\right]$$

where, Hr is the reference dose, Hm is the laboratory measure dose and Ho is the detection limits (0,2 mSv).

The performance of TLD and film systems over the programme is shown in Figure 1. Figure 2 shows the laboratory percentage for the cases of TLD and film systems, which have a good performance response (90% of their results inside the acceptance curve).
Figure 3 shows the performance percentage for both systems in terms of the energy quality. The angular response study showed that the 60% of the TLD and the 70% of the film results were correct.

CONCLUSIONS

The intercomparison programme was carried out by most of the processors. However, only 41% of the laboratories have completed the programme. All the laboratories that participated also answered the *ad hoc* questionnaire providing general information about the working methodology. From them, the following information was obtained:

- The 56% of the laboratories used film and the 44% used TLD.
- Related to the quantities informed, at the end of the sixth exercise, the 33% of the results have been expressed in terms of personal dose equivalent. The 50% expressed them in terms of dose and the unit used was mSv. Only one expressed them in term of deep dose (mSv) and another one expressed in dose (mrem).
- The dosemeter complexity does not improve the result accuracy. The capability of the dosemeter to discriminate energies is not related to the resultant accuracy.
- Only the 60% of the processors did their calibration at CRRD facility that offers it as a paid service.
- Laboratories that changed the dosimetry manager many times have a wide fluctuation in the results over the years.
- Laboratories whose staff was trained by CNEA have obtained the best performance

The results show a better performance of TLD systems than film systems, throughout the program, film systems were improving their results.

The 92% of the processors can obtain a detection limit less than (or equal) 0.2 mSv that fit quite well with the acceptance curve. The high dose is well measurement by the 70% TLD laboratories and the 50% of film laboratories that because they did not work with double film.

The dosemeters vary widely in the energy response characteristics, depending on design and evaluation techniques. The accurate measurements in the mixed energy field presented more difficult for film than TLD.

The experience gained during these years provides a sound basis for developing an action plan aimed at improving the qualities of the personnel individual monitoring systems in Argentina.

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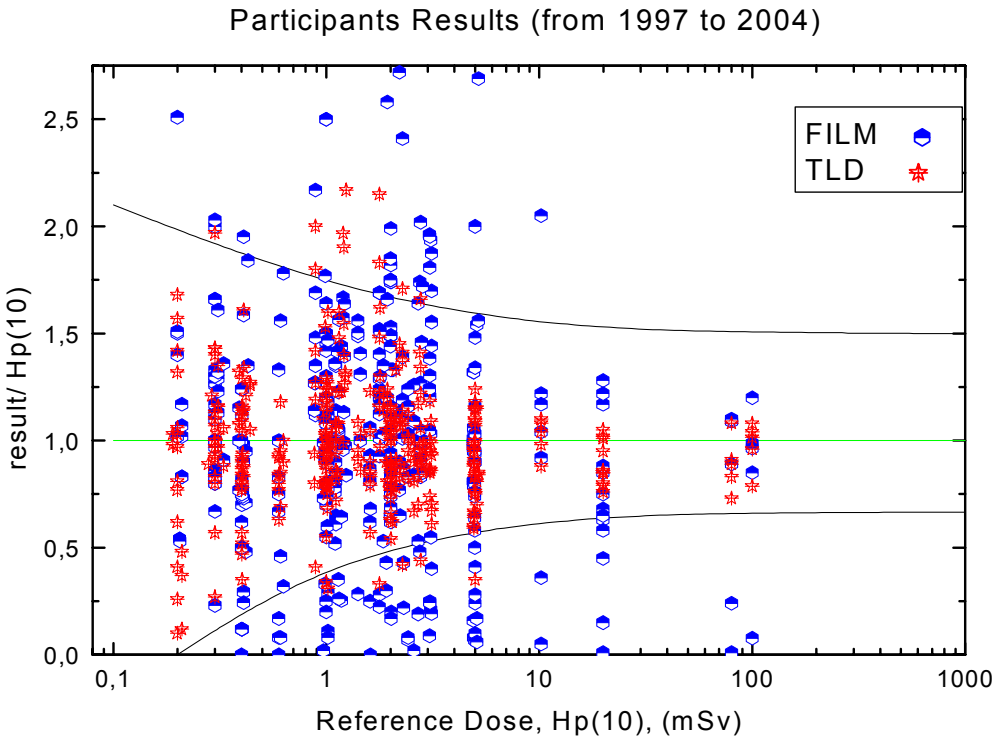

Figure 1. TLD and Film Dosimetry System Performance Comparison

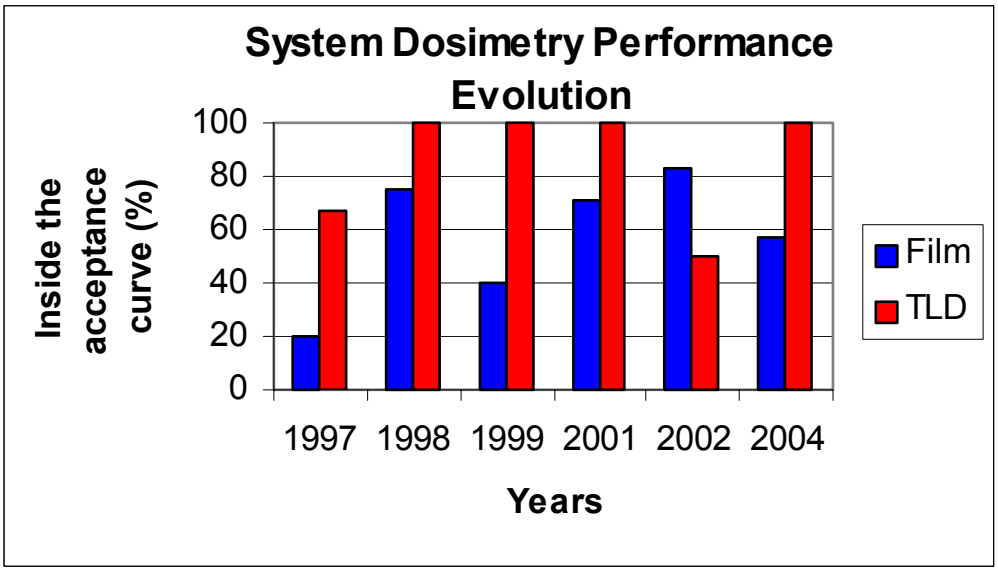

Figure 2. Laboratories results with 90% inside the trumpet curve

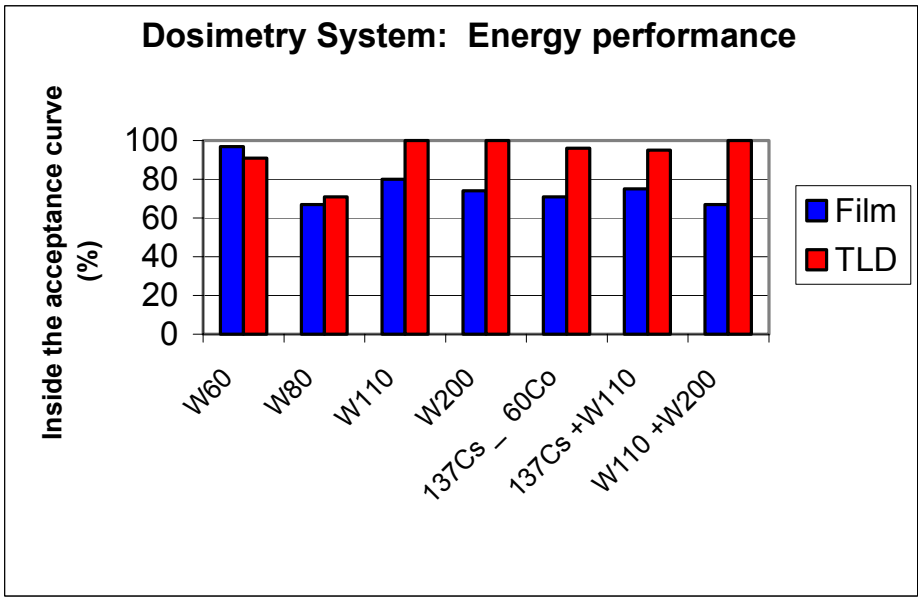

Figure 3. Energy Performance for both detection systems

Initial Actions to Control Emergency Situations Involving Missing or Stolen Radioactive Sources

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INITIAL ACTIONS TO CONTROL EMERGENCY SITUATIONS INVOLVING MISSING OR STOLEN RADIOACTIVE SOURCES

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Abstract

The regulatory standards in Argentina establish that it is the licensee's responsibility to respond first in a radiological emergency. To obtain a license from the Regulatory Authority the operators of radioactive installations or mobile equipment with radioactive sources must demonstrate that they have the skills and the resources to respond immediately as soon as an emergency situation arises. The Regulatory Authority has its own Emergency Intervention Response System (SIER). This system is prepared to assist licensees and to intervene in cases where sources are not associated with an authorized user, such as transport accidents or in cases associated to orphan sources. Many emergency situations arise when a dangerous source is misplaced or stolen, a situation that is more feasible in mobile radioactive devices, like those of industrial gammagraphy.

Both the SIER and the Licensees in Argentina apply without any delay similar first actions, including actions to recover the radiological control, recover the misplaced sources and mitigate or repair consequences in persons and environment. The present work describes the initial actions planed for alert and notification (local, national and international), coordinated involvement of other authorities (security forces, justice) and informing the public through the media.

Two examples (Neuquén and Río Grande) are presented to demonstrate the effectiveness of the methods applied which could be useful to other radiological intervening organizations in other countries.

Keywords: Emergency Response, Radioactive Sources, Orphan Sources

1. Users as the first prepared responders

The Argentinean users of radioactive sources in medical and industrial applications shall demonstrate to the Nuclear Regulatory Authority their technical qualifications regarding practices and their technical and operative capability to apply the principles of the radiological protection, for workers, the public and the environmental. This protection involves not only normal operation but also emergency situations due to incidents and accidents during the use, storage or transport of the radioactive sources.
The training courses and the practical activities of training and periodic re-training always include preparation and response for radiological emergencies topics.
By this means, the well-qualified licensee is the first protection barrier in front of an accidental situation. This is the requirement included in the regulations and the extent of a licensee's responsibility is perfectly defined.

2. The SIER as a back-up

The work that the Nuclear Regulatory Authority (NRA) carries out related to accidents with radiation and radioactive material, in those that could affect workers, members of the public and the environment, is fundamentally to set up a policy of prevention.

With this purpose, in the evaluations of safety previous to authorization of each practice the risk of accidental situations is considered.
For each possible accidental situation, the potential consequences are evaluated and, if necessary, additional measures are required to be put in place to ensure that the probability of occurrence and the seriousness of the consequences are properly limited. The result is that, the likelihood that someone is affected by an accident with radiation is in fact very low.
Additionally, as we mentioned, the operator is required to have enough qualification and training so that, if detecting any anomaly that could lead to an accident in advance, they are ready to apply the procedures of emergency in order to avoid or to minimize their consequences.
The responsible party for the first response is the licensee, who has the skills for making it. He is also required to notify immediately to the NRA any abnormal situation. Having this notification, the NRA performs an independent assessment of the situation and the applied countermeasures and, consequently, recommends or imposes modifications in the procedures in order to avoid similar situations in the future. In all the cases a final independent assessment is carried out and the incident is registered.
The NRA in Argentina has its own response system for radiological accidents: the Radiological Emergency Intervention System (from Spanish: SIER). This system stays operative 24 hours per day during the year. The SIER intervenes supporting those users that are involved in accidents. Furthermore, the SIER is the responding organization in those events that do not have a responsible user with knowledge in radiological safety, e.g.: accidents that take place from the unexpected appearance of radioactive sources outside of authorized facilities (products of illicit activity, accidents during the transport of radioactive material, etc.). For these cases Argentina has the SIER like the national competent authority.
In this case, the request of the SIER is done, in most cases, by Security Forces, the Fireman Brigades, the Civil Defense and sometimes members of the public.
The SIER carries out ten interventions per year on average, the majority without radiological consequences, thanks to a quick and effective response capacity.
At the same time, the SIER is responsible for maintaining operative and coordinating a National Response Medical Network for attention of people over-exposed. This Network is articulated in 4 levels of growing complexity:
- Level 1: involve the medical services of facilities and the emergency systems of medical response at the site of event (pre-hospital response).
- Level 2: involve local hospitals with medium-level capability to the site of event.
- Level 3: involve institutions with high capability to attend patients with more severe cases.
- Level 4: involve the response of international medical cooperation. The NRA's SIER is a member of the Radiological Emergencies Preparing Assistance Network of World Health Organization (WHO/REMPAN), for requesting and offering advice and assistance in accidental situations.
Inside this Network, the NRA organizes meetings for training, makes recommendations, protocols and guides, and maintains connection with international centers in radio-pathology.

3. Emergencies with sources outside the regulatory control

A very special case that has been producing serious exposures to radiation to persons in the entire world is the theft or loss of dangerous radioactive sources. In these cases the source is out of control and people may not know their exact danger. The affected licensee is responsible to apply the first actions: to notify the police, the judicial authorities, the NRA and other competent authorities that may be involved. A large multidisciplinary working team is formed. The user should put all the resources of his company to try to find the source before it produces undesirable effects. The NRA encourages and supervises this work, but the user is always identified as primarily responsible because the user is the generator of the risk. Nevertheless, the regulatory authority and the other authorities contribute actively in the implementation of the recovery actions.
The judicial authorities, when alerted, use the NRA as the technical expert to assist in the definition of judicial measures to facilitate the recovery. These measures run from an alert to all police forces in the country, up to forced inspections of houses or other buildings. All the security forces work under the coordination of the judicial authorities but with the SIER as the technical advisor and director of the radiological countermeasures.

A method applied in Argentina is the broadcast of messages to the population. These messages are the licensee's responsibility and enforced by the NRA. Nevertheless the Nuclear Regulatory Authority supervises and controls permanently the subject, scope and information campaign. Far from making them fall into panic, these messages are able to put the population and the security forces in alert. In addition, the commerce in the black market is prevented most of the times as the people realize that these devices with radioactive sources represent a danger and that the police are trying to find it.
For the authors of illicit activities, by means of public messages they are alerted that material that they have stolen doesn't have a significant commercial value, because nobody can buy it or use it without the due authorization and that, if it is manipulated incorrectly, they can put in hazard their lives and their family. Always, the messages mention that, managed correctly and inside their corresponding original containers, the materials have low risk, giving this a way to escape from this situation. The experience in Argentina shows that generally they give back the source in relatively safe manner, in their original containers.

A measure of high impact to achieve the recovery is the public announcement of an imminent aerial survey. The announcement of this aerial or terrestrial survey makes the authors of the illegal activity consider that they will be identified. In all cases applied in Argentina the burglars had returned the sources immediately, making them appear one way or another without putting in risk the population or the environment.

In our experience, in all the situations where thefts took place, the thieves ignored the technical characteristics of what they were taking. They carried it out only guided by the instinct that it was something valuable and peculiar. The public messages always helped to clarify what a radioactive source is. In many cases, the thieves returned the sources saying that they had found it and that they want to give it back to help with the police. In no case did these messages generate adverse radiological situations.

In all these cases, the licensees, the competent authorities including the NRA, the police and the security forces intervened in a coordinated manner. The judicial authority exercised leadership, however the regulatory authority permanently advised it in the aspects of radiological safety. The coordinated work of all these authorities and forces is facilitated by means of a permanent cooperation and information exchange program carried out by the NRA.

4. Examples

Radiological incident in Neuquén

On Saturday, June 9, 2002, the Radiological Emergencies Intervention System (SIER) received a call from a company that performs industrial radiography, stating that gammagraphy equipment with a source of Ir-192 with 53 Ci of activity was stolen at Neuquén city, in the south west of the Argentinean territory (about 1100 km from Buenos Aires city).

Actions

- The Emergency Response Center was activated.
- The company was required to report the situation to Federal Justice, and to begin a media campaign to alert the population about the dangers of manipulation of the lost equipment.
- A primary intervention group of specialists was sent in a rented private plane to intervene and to steer the response to the emergency in place and make contact with the Federal Police in Neuquén and the Federal Judge.
- The licensee and the Nuclear Regulatory Authority made statements that were distributed locally and by the national media.
- The Emergency Response Center is asked to organize an aerial survey in Neuquén city, in coordination with the SIFEM (Federal Emergencies System)

Results

- The intervening group from the SIER received from the Federal Judge a description of the facts in the presence of the other intervening organizations (Instruction Judge, Head of Neuquén Police, Provincial Environment Secretary, Head of Firemen Brigades, and National Border Guard).
- The company notified that they have received a call and that someone had founded the equipment and had it in his house in a safe condition waiting for the police.
- The Judge states that after the company published warning messages in the Television and the news papers to give alert on the event, a family notified that they had the equipment in the back yard of their house.
- It was verified that the equipment was safe.
- Cytogenetic dosimetry was performed on the members of the family and no indications of high radiation doses were detected.
- Investigation showed that the grand father of the family was responsible for the theft, and that after many messages his family encouraged him to call the company to say they had the equipment in the house. No one was convicted.

Radiological incident in Río Grande, Tierra del Fuego

The 19th of November 2003 the SIER received a phone call from a gammagraphy company 3000 km from Buenos Aires) announcing the theft of the equipment with a source of Ir-192 with an activity 6,5 Ci.

Actions:

- The Emergency Response Center was activated.
- The licensee informed the local Delegation of the Federal Police and the Provincial Police and to start a media campaign immediately to alert the population about the dangers of improperly handling the equipment.
- The SIER sent an intervening group by plane to Río Grande.
- The SIER warned the National Border Police (Gendarmería), because this region is close to the frontier of Chile.
- National Border Police contacted customs officers in the frontier and they acted in coordination.
- The SIER warned the Federal Police, the Provincial Tribunal and the National Coastguards.
- The Chilean Competent Authority was notified in the framework of the Early Notification Convention. Also the IAEA was notified in this framework.
- The Licensee and the SIER team started a media campaign, through press, local television and radio, including media conferences.
- A terrestrial survey with vehicles and radiation detectors was initiated.
- Specific intelligence tasks were coordinated with the Brigade of Investigations of the Judicial Police to obtain more information.
- Leaflets with information on radiation injuries were distributed at hospitals.
- On the 22nd of November an aerial survey was announced to the population for the following day by local media and two helicopters form the Provincial Government arrived in the city.

Results

- Hundreds of calls with information about the equipment were received and all the cases were investigated.
- The 23rd of November a phone call was received at 03:30 a.m. by the Judicial Police, saying that the equipment had been seen in a neighborhood.
- The SIER intervening group, together with police, arrived at 4:10 a.m. to the neighborhood and found the equipment in the street intact with its source in a safe condition.

5. Conclusions

- The authorized user of radioactive source is the first responder and is prepared to do so.
- The regulatory authority has the SIER for user's support and cases with orphan sources or accidents in public areas.
- For lost or stolen sources the coordinated actions between the radiological experts, the justice and the security forces is the best response
- Media campaigns and announcements with information of the risk involved, the first actions to avoid radiological consequences and the planned recovery actions, through press, local television and radio, including media conferences always have given positive results.

A Web Portal for Use in Emergencies Involving Radioactive Sources

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A WEB PORTAL FOR USE IN EMERGENCIES INVOLVING RADIOACTIVE SOURCES

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ABSTRACT

Incidents involving dangerous radioactive sources can require a range of emergency response activities. Many IAEA Member States have highly-capable emergency response organizations for responding to such emergencies, but some Member States do not. In the general context of the International Action Plan for strengthening the international preparedness and the response system for nuclear and radiological emergencies, which is under development by the National Competent Authorities Coordinating Group (NCACG) and the IAEA Secretariat, an Assistance Working Group (AWG) has been established to work on the development of effective, efficient and compatible arrangements whereby Member States can obtain relevant and adequate assistance. The Expert Group B.2 of the AWG has been commissioned to examine the case of emergencies involving lost, damaged, stolen or discovered radioactive sources. The Expert Group is proposing the development of a web-based software system to provide guidance to responsible organizations during these events. The software system, called Emergency Response Web Portal for the Management of Radiological Events Involving Radioactive Sources (Web Portal) will include an expert system and a toolbox. The expert system will examine inputs provided by the affected State during an emergency to suggest appropriate responses and the types of assistance that may be available. The expert system would also provide access to modules in the toolbox, which could include databases, sections of relevant regulatory and technical documents, and lessons learned. The Web Portal would be useful for planning purposes and for use in actual emergencies.

Keywords: Emergency Response, Radioactive Source, International Assistance

1 Introduction

There has been a growing realization since the 1986 Chernobyl accident that radiological emergencies generate concern and attention that are not limited to the borders of the immediately-affected State. The public perception of risks associated with radiological accidents has also been exaggerated compared to other hazards. Thus two conventions were drafted in 1986, and have been signed by a large number of IAEA Member States. These two conventions are the Convention on Early Notification of a Nuclear Accident, and the Convention on Assistance in Case of a Nuclear Accident or Radiological Emergency [1]. The conventions have guided international cooperation in responding to emergency events. Under these conventions competent authorities are defined for Member States, and these competent authorities are responsible for issuing and receiving information relating to a nuclear accident.

The Second Meeting of the Competent Authorities, convened by the IAEA Secretariat in Vienna in June 2003, made progress on the issue of rendering international assistance during a radiological or nuclear emergency by authorizing the formation of a National Competent Authorities Coordinating Group (NCACG), and charged this group with developing an action plan that would address improvements to the international systems for emergency preparedness and response. The NCACG prepared a draft International Action Plan for Strengthening the International Preparedness and Response System for Nuclear and

Radiological Emergencies[1] in 2004 and an Assistance Working Group (AWG) has been established to carry out the activities specified in the Action Plan.

The AWG has a goal of developing effective, efficient and compatible arrangements whereby Member States can obtain relevant and adequate assistance in the event of a nuclear or radiological emergency. A number of Expert Groups were formed under the AWG to implement specific portions of the Action Plan. Expert Group B.2 has the responsibility to develop arrangements for response to situations involving lost, stolen, damaged or discovered dangerous sources.

2 Expert Group B.2

Radioactive sources, both sealed and unsealed, when designed, managed and used in accordance with existing national and international standards are safe tools whose use results in substantial benefits to medicine, industry and agriculture. Unfortunately, there have been worldwide a relatively large number of radiological accidents reported over the last fifty years involving unsafe, abandoned, lost or uncontrolled radiation sources. A number of these accidents have resulted in death, serious injury, environmental damage and substantial financial liabilities; and have required substantial cost and effort for remediation. Radiological accidents not only include exposures to persons and environmental contamination, but also orphaned sources (sources outside of regulatory control or lost and abandoned altogether) finding their way into metal scrap destined for recycling. In addition, the spectre of terrorism has highlighted the threat posed by the use of radioactive sources in radiological dispersion devices.

In the well-documented radiological accident involving loss of control over a teletherapy ^{137}Cs radioactive source in Goiânia Brazil in 1987, a total 112,800 individuals were surveyed and 249 persons were found to be contaminated with radioactive material. To indicate the complicated nature of the public health crisis that followed the incident, more than 5% of the surveyed persons exhibited some signs and symptoms of acute radiation sickness although they had not been exposed [2,3].

The focus of Expert Group B.2 is on health and safety concerns regarding radioactive sources. Security aspects of these events are not specifically in the Expert Group's objectives. The group's objective is to improve international capabilities for responding to situations involving potentially dangerous radioactive sources by providing Member States with practical guidance for establishing international arrangements for providing assistance to ensure effective and efficient evaluation and proper handling of lost, stolen, damaged or discovered sources to protect public health and safety. Recommendations developed during this work will take into account the basic principles of the Code of Conduct on the Safety and Security of Radioactive Sources[2].

For the purpose of this work, two documents are relevant to describing dangerous sources, TECDOC-1344 and the HASS. The European Union (EU) published a Council Directive on the control of high-activity sealed radioactive sources and orphan sources ("HASS") [4]. The HASS presents requirements for EU members for the use and control of radioactive sources. As part of this directive, sources of interest are identified, using a criterion that was developed by the IAEA for safe transport of sources [5]. TECDOC-1344 [6] uses a different method to categorize radioactive sources, using a criterion that is based on health effects which can be observed soon after exposure. It bases the categorization on a "dangerous" activity, which would be sufficient to cause a deterministic health effect in a person by a reasonable method of exposure to the source's radiation. Under this convention, Categories 1, 2 and 3 represent sources that emit radiation sufficiently intense to cause serious health effects in an exposed person. Thus sources in Categories 1, 2 and 3 are the focus of this Expert Group's recommendations.

[1] http://www-ns.iaea.org/downloads/rw/action-plans/ers-action-plan.pdf
[2] http://www-pub.iaea.org/MTCD/publications/PDF/Code-2004_web.pdf

3 Web Portal

Expert Group B.2 has developed a concept for a software system, the Emergency Response Web Portal for the Management of Radiological Events Involving Radioactive Sources, that will assist Member States in the response to an event involving a lost, stolen, damaged or discovered radioactive source. There would be two modes of usage planned for this software system: a planning mode and a response mode. Organizations with responsibility for emergency response in Member States could run the software in the planning and development mode, simulating potential incidents, to prepare their organizations for response to these events and develop necessary capabilities. In the event of an actual incident, the software could be run by emergency organizations in the incident management mode to gather advice on necessary response activities, to gain easy access to relevant technical information and regulatory documents, and to determine the availability of international assistance that could be used during the event.

The web portal would have two basic components, an Expert System component that would acquire user input, work through a decision tree process, and generate output to the user; and a Tool Box that would contain useful handbooks and calculation tools, plus a library of technical and regulatory documents. The general approach of the web portal is illustrated in Figure 1.

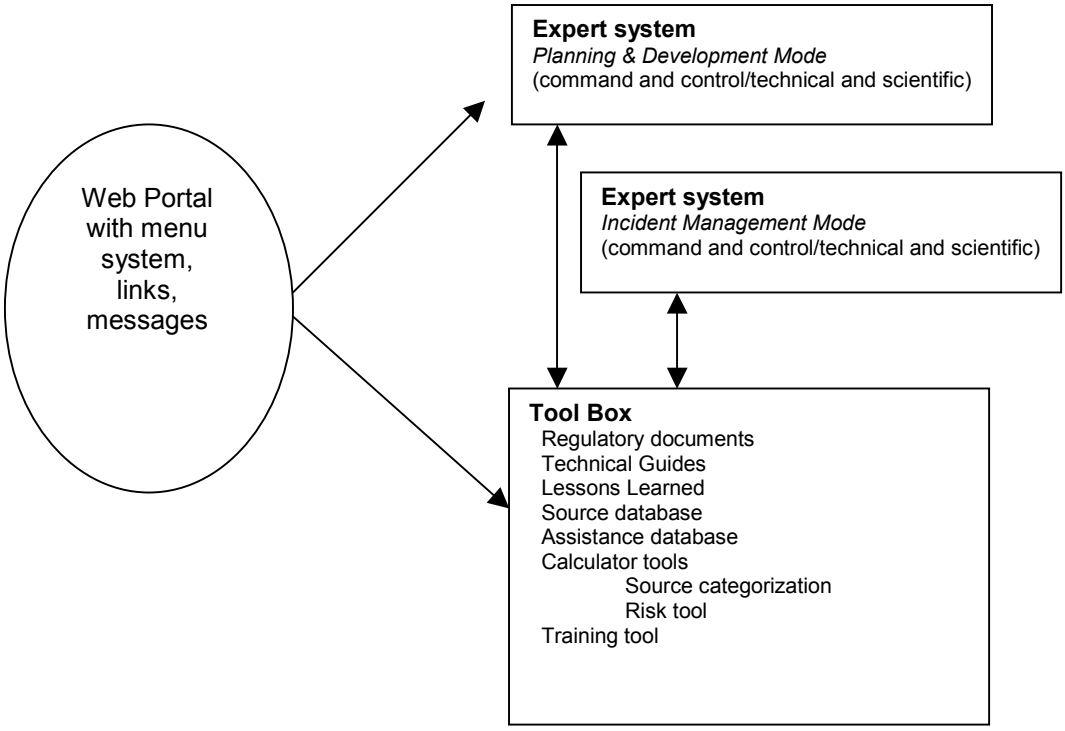

Figure 1: General approach of the web portal

3.1 Expert System

The expert system would accept input from the user, work through a decision tree process to classify the incident and gather relevant data to provide to the user, then present output to the user in the form of recommendations and links to useful tools and documents.

In the input section, the user would provide information that can be used to categorize the incident. This could include information about the radioactive source that is the subject of the incident, and whether it is lost, stolen, damaged or discovered. The component radionuclides and activities (if known) would be stated, the ownership and recent custodianship of the source,

any geographical information, and timing of the significant events would all be input. Some of this input may not be in the format of traditional text and numerical user input, but may consist of photographs, maps or other digital imagery.

After all input has been collected by the Expert System, a decision tree would operate to classify and categorize the event, then gather information that would be useful to the user. The decision-making process will have to be built into the software based on expert knowledge and a study of previous events and associated lessons learned.

After the Expert System has completed the decision tree process, it would then produce output. First would be its analysis and categorization of the incident, including an assessment of potential health and safety risks to the general population and emergency responders. Appropriate actions that should be taken in response to the incident would then be presented. Capabilities required for this response that may not reside in the affected State would be identified, and nearby States and organizations that possess these capabilities would be listed. Relevant sections of documents that pertain to this incident would be linked into the output, in a prioritized presentation, so the user could refer to these as desired. Lessons-learned documents can also be presented in this list. Calculation tools that would be useful for additional simulations would also be available.

An example pathway for a segment of the Expert System's logic is shown in Figure 2. Notice that several pathways refer to numbered circles that indicate sections of the software that are outside of this logic segment.

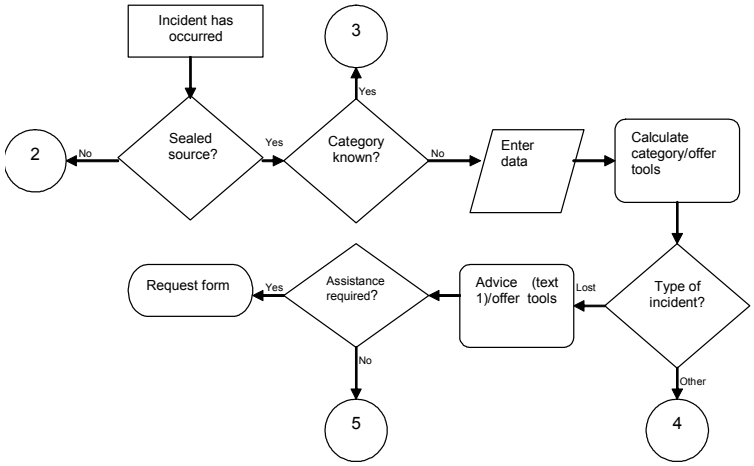

Figure 2: Example pathway for segment of Expert System logic

3.2 Toolbox

The Toolbox would contain additional resources that would be helpful to the user in the specific incident. The resources in the toolbox could be accessed by the user directly, without going through the Expert System, but in most Web Portal usage, the resources in the tool box would be retrieved by the Expert System.

The Toolbox would include a number of documents, including technical guides and regulatory documents. It would also include a collection of documents describing relevant incidents, with "lessons learned" to guide response organizations in their planning and actual response.

The Toolbox would also include relevant databases. A database of radioactive sources would be included, and this could be expanded to include the sources registered in the local Member State. A database of capabilities that reside in States in the region, and are available to assist nearby States during emergencies, may be included in the Toolbox.

There would also be calculational tools which are helpful in these incidents. A tool for determining the categorization of a source according to TECDOC-1344 will be included in the toolbox. There may also be a calculation tool that relates source quantities to radiation doses under several well-defined scenarios and relates these doses to risk of human health effects.

The Toolbox could also contain items such as training information and sample press releases. All contents of the Toolbox will be organized so that the Expert System could readily access relevant information for a given incident and easily partition useful information from extraneous information for the given incident.

4 Discussion and Conclusions

Expert Group B.2 will be designing the specifications for a demonstration version of the Web Portal. This prototype version will include functionality for a single limited set of incident conditions. The prototype version will demonstrate the basic concept of the software and test the usefulness of the overall package.

Expert Group B.2 has developed the concept of the Web Portal as an efficient method to provide uniform guidance to Member States for response to incidents involving lost, stolen, damaged or discovered radiological sources that are potentially dangerous. This software will help the AWG fulfill its commitments to the NCACG Action Plan.

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Prevention as the Main Objective for Regulatory Practices Related to Research Reactors

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PREVENTION AS THE MAIN OBJECTIVE FOR REGULATORY PRACTICES RELATED TO RESEARCH REACTORS

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1.- Introduction

In Argentina research reactors and critical assemblies are used with different and varied purposes, ranging between radioisotope production, scientific research and teaching.
The license holders characteristics are diverse as well, considering the National Atomic Energy Commission (CNEA) and National Universities as typical examples.

The following Table summarises some data from research reactors and critical assemblies in Argentina:

Reactor	Licence Holder	Type	Power	FE Type	Enrich.	Shut down systems	Excess reactivity
RA-0	Cordoba University	CA *	1 W	Cylindrical rods	< 20%	Absorbing rods Empty Moderator	0.4 $
RA-1	CNEA	RR**	40 kW	Cylindrical rods	< 20%	Absorbing rods	1.2 $
RA-3	CNEA	RR	10 MW	MTR plates	< 20%	Absorbing rods	~ 8 $
RA-4	Rosario University	CA	1 W	Homogeneous disks	< 20%	Absorbing rods Separation	0.4 $
RA-6	CNEA	RR	500 kW	MTR plates	90%	Absorbing rods	~ 2.5 $
RA-8	CNEA	CA	10 W	Cylindrical rods	< 3.4%	Absorbing rods Empty Moderator	

* Critical Assembly ** Research Reactor

Taking into account such a varied scenario and having in mind the potential risk associated to the operation of each installation, the strategy used for regulatory control is mainly based on PREVENTION.

2.- PREVENTION, the most important regulatory strategy

From the regulatory point of view, PREVENTION is achieved with the help of:

- *First of all, An adequate Regulatory Frame*
- *Second, an ppropriate regulatory tools*
- Third, a *pro-active attitude towards safety*
- *Fourth, an qualified staff for the correct fulfilment of tasks assigned*
- *And fifth, An adequate organisation*

2.1.- Regulatory Framework

Regarding the Regulatory Frame, there are defined responsibilities and functions of the Nuclear Regulatory Authority and of the Licence Holder Institutions of nuclear reactors through National Acts.

The responsibilities and functions of the mentioned institutions are compatible with those recommended in the Behaviour Code for Research Reactors.

As refers to the regulatory frame related to nuclear reactors, the Regulatory Body has the following responsibilities on:

- Issue Standards and Guides
- License installations
- License personnel
- Apply the sanctions and fines regime

2.1.1.- Related to the Issue of Standards and Guides, there are specific Regulatory Standards on application for research reactors as well as for critical assemblies.

In the elaboration process for standards and guides, international recommendations are taken into account together with national and international operation experience.

A consultation mechanism has been implemented through which the opinion of the addressees and the public is requested prior to the formal approval of new standards or modification of already existing ones. It is not a binding consultation, and its purpose is to give the opportunity of giving an opinion about technical aspects and about the consequences that could derive from the implementation of the new or modified standard.

The Argentine regulations establish that a Licence Holder Institutions should exist as well as a Primary Responsible, who is the Reactor Head. Besides, the standards require that two Committees should exist in order to analyse radiological and nuclear safety related aspects.

- Technical Review Committee, to assess the Licence Holder Institution. It should be composed of installation independent specialists, who should be experienced professionals in nuclear reactor operation and accepted by the Regulatory Body.

  The Committee should meet periodically in order to review the development of reactor operation from the point of view of safety and people protection. It should include the analysis of operation incidents and any other relevant event that may have happened as well as the assessment of proposals made by the Primary Responsible or the Licence Holder about modifications to the installation design or new experiments or exposures involving the reactor.

- The Internal Safety Committee, to assess the Primary Responsible, constituted by personnel belonging to the installation as well as to other organisations.

  The Committee members should be named by the Primary Responsible and it should be constituted by no more than 50% members of the installation, being the rest of its members part of other sections of the organisation or independent experts. This Committee advises the Primary Responsible about radiological and nuclear safety aspects appearing during the reactor operation.

The Argentine Standards have been broadly applied in the licensing of research reactors, to authorise significant modifications carried out in the RA-3 reactor during its updating and power upgrading and for the authorisation of new experiments such as U-Si prototypes FE. Moreover, the Standards have been a binding tool in the design of reactors built by Argentine companies for other countries: Peru, Algeria, Egypt and Australia, in which the Argentine Regulatory Body at least gave its opinion concerning design licensability. In some of these cases, the Argentinean criteria for personnel licensing were also used, and the Regulatory Body helped during the process.

2.1.2.- Regarding the License of installations, the regulatory process considers the issue of Construction, Start up, Operation and Decommissioning Licenses. No Special License for the site is required, as the interaction installation-site is evaluated as a whole when issuing the Construction License.

For nuclear reactors the Construction License has a maximum validity of five years and as a condition prior to its renewal, the Licence Holder must submit to the Regulatory Body an integral assessment of the installation safety.

As a part of the process of licensing new installations, the Regulatory Body is now considering the application of a mechanism of non-binding public consultation.

2.1.3.- Regarding to the License personnel process, The Regulatory Body issues two conceptually different types of documents to license personnel working at the installation:

- *Individual License*: It is a permanent certificate that recognises the technical-scientific qualification needed to exercise a particular function in the Organisation Chart for a certain type of installation (e. g. the Critical Assembly Reactor Head). In order to obtain it, it is necessary to demonstrate a basic qualification, a specific one and working experience.
- *Specific Authorisation*: It is a renewable certificate having a maximum validity of two years, which enables a person to exercise a certain function at a particular installation. In order to obtain it, it is necessary for the applicant to have the corresponding Individual License for the position and his specific qualifications, on the job training, re-training and psycho- physical aptitude are evaluated.

2.1.4.- In case of non-fulfilment of what is established in the mandatory documentation (License, standards, Safety Assessment Report, the Regulatory Body can apply a Sanctions Regime, which considers, depending on the seriousness of the case, warnings, fines, a License or a Specific Authorisation suspension, or even its revocation.

The Sanctions Regime acts as the last step in the safety chain. Moreover, if the system is effective and if the License Holders fully exercise their responsibilities, a sanction application should only occur exceptionally. Otherwise, it should indicate a poor regulatory behaviour. In that sense, it should not be a formal function of the Regulatory Body to help the Primary Responsible and License Holders in being conscious of their responsibilities concerning safety and each time more identified with the Safety Culture application.

2.2.- Regarding the Regulatory control, it is mainly carried out through inspections, audits and safety assessments. Besides, a strict control of dose to workers, radioactive material discharge to the environment, dose to the public and environment surveillance in the surroundings of nuclear installations are also performed.

There are three types of inspections and audits:

- Routine inspections that consider subjects such as radiological protection, maintenance and operation.
- Non routinely inspections that take place when certain events occur; to observe start up tests associated to modifications to the installation or new experiments, etc.
- Special inspections that take place at the beginning of the annual operation period after a programmed maintenance outage. Their purpose is to verify the general maintenance tasks carried out, safety system functioning and "operable core" conditions.

The goal of installation inspections and audits is to verify that:

- Conditions established in mandatory documentation are fulfilled.
- Design safety functions for installation structures, components and systems are preserved.
- At every moment, necessary staff with the required qualification to carry out regulatory control tasks is available.
- The principles of Safety Culture are applied satisfactorily.
- Corrective actions to remove and avoid new occurrence of deficiencies and abnormal conditions detected are performed.

At the beginning of each year a program of routine inspections and audits is planned. The scope of such program as well as the inspection frequency consider not only the postulated potential risk but also compliance with the above-mentioned goal.

For the program implementation several methodologies are used that may be grouped as follows:

- Direct observation of task execution.
- Interviews with personnel.
- Verification and control of procedures, records and documentation.
- Verification of personnel specific training.
- "In situ" verification , if possible, of set points and safety system actuation.
- Whenever it is possible, take samples, tests, verifications and independent measurements (particularly in the radiological area).

For each installation there is a "check-list" to help in the preparation and execution of inspection activities.

Two types of safety assessments are performed:

- Routine evaluations, effluent discharge, waste management, dose to workers, committee safety reports, etc.).
- Non-routinely evaluations relevant events, installation significant modifications or new experiments, safety assessment report updating, core management, exercises for the emergency plan application, compliance with regulatory requirements, etc).

The Regulatory Body staff assigned to research reactor control is supported by specialists in subjects such as: Radiological protection, activity measurement, neutronics, thermal hydraulics, mechanics, electric aspects, materials, PSA, I&C and operation safety (event assessment, safety culture, quality assurance) also used for nuclear power plant control.

If unfavourable trends or deviation from normal operation conditions are detected during regulatory control tasks, preventive or corrective actions can be applied such as issuing of regulatory requirements.

As regards environmental surveillance, the regulatory actions consist of monitoring of nuclear installation surroundings, taking representative samples from the different regions of the radionuclide transfer environmental matrix, and evaluating the impact to the environment of liquid and gaseous discharge.

With such purpose water samples, sediments, plankton, fish, drinking water, grass, regionally produced food, etc, are taken.

The radiological impact of results obtained from environment monitoring is determined by dose to the public estimation. Doses are to an average individual belonging to the critical group, with the hypothesis that the food consumed at the site is of local origin.

The Regulatory Body has a Nuclear Emergency Intervention System (SIEN), and a Radiological Emergency Intervention System (SIER) to cope with radiological emergencies in minor installations and practices, or emergencies that could involve the public. They are linked to a Federal Emergency System and they also advise public authorities as well as users. The installations mast do a simulation emergency exercise once a year.

2.3.- The existence of a pro-active attitude towards safety is one strong aspects of the regulatory action. Such attitude should be shown for example, in:

- Continuous review of international recommendations and standards in force.
- Frequent presence of inspectors at installations.
- Active participation in examination boards for personnel licensing and re-training.
- Application of quality and continuous improvement aspects in regulatory tasks.
- Performance of periodic meetings with License Holders or their representatives, with open discussions about existing problems as the adequate means for achieving improvement in both installations and organisations.
- Support for the application of the Behaviour Code for Research Reactors.

2.4.- In order to fulfil the regulatory mission adequately, it is a necessary condition to preserve the staff technical qualification. This goal may be achieved by means of a continuous training program including teaching activities. The main activities related to training are:

- Post-grade course attendance on radiological and nuclear safety for new professional and technical personnel.
- Attendance to specific training courses.
- Participation in international and national meetings and workshops.
- Collaboration in international and national training activities.
- Collaboration with other regulatory bodies in specific subjects.
- Participation in document elaboration in radiological and nuclear safety areas, particularly in IAEA technical documents.

2.5.- The organisation should enable transparency and traceability in the regulatory task, where individual and collective responsibilities should be clearly established in order to carry out the necessary control; mechanisms to find and correct deficiencies should also exist.

Globally, this means that it is necessary to plan, perform and control activities related to the regulatory action in nuclear reactors, and plan the required economic and human resources.

In particular, tasks should be carried out according to specific procedures. The Regulatory Body has procedures for inspections, audits and assessments, event communication, regulatory control during the application of emergency plan exercises and issuing of regulatory requirements. Annex II lists applicable procedures for research reactors.

The nomination of Process Responsible staff was created as a mechanism of finding and correcting deficiencies.

2.6.- Improvements opportunities. In that sense the regulatory authority is working to get a better

- Actuation of Technical Review Committees. Some times It was considered only formally and not as an independent review mechanism of the operation organisation.
- Also is necessary improve the Emergency Plans at Atomic Centres, and its interrelation with the Emergency Plan of the Installation itself
- We are also working on new Standards on Decommissioning, to do them more specific.
- Peer Review. Installations reluctant to carry it out.

ANNEX I

List of Argentine Standards and Guides for research reactors and critical assemblies.

Regulatory Standards

Code	Standard Title	Review
AR 0.0.1.	Licensing of Class 1 installations	Rev. 2
AR 0.11.1.	Licensing of Class 1 installation personnel	Rev. 3
AR 0.11.2.	Requirements of psycho-physical aptitude for specific authorisations	Rev. 2
AR 0.11.3.	Personnel re-training for Class 1 installations	Rev. 1
AR 4.1.1.	Work exposure in research reactors	Rev. 0
AR 4.1.2.	Radioactive effluent limits in research reactors	Rev. 1
AR 4.1.3.	Radiological criteria referred to accidents in research reactors	Rev. 2
AR 4.2.1.	Critical assemblies design	Rev. 1
AR 4.2.2.	Research reactors design	Rev. 1
AR 4.2.3.	Safety against fire in research reactors.	Rev. 2
AR 4.5.1.	Electric Power Supply system design in research reactors	Rev. 1
AR 4.7.1.	Time table of documentation to be submitted before operation of a research reactor	Rev. 1
AR 4.7.2.	Time table of documentation to be submitted before operation of a critical assembly	Rev. 0
AR 4.8.1.	Preliminary tests and start up of critical assemblies	Rev. 1
AR 4.8.2.	Preliminary tests and start up of research reactors	Rev. 1
AR 4.9.1.	Critical assemblies operation	Rev. 1
AR 4.9.2.	Research reactors operation	Rev. 2
AR 10.1.1.	Radiological Safety Basic Standard	Rev. 3
AR 10.12.1.	Radioactive waste management	Rev. 1
AR 10.13.1.	Radioactive materials transport	Rev. 1

ANNEX II

List of applicable procedures for the regulatory control of research reactors and critical assemblies.

Code	Review	Document title
G-0XX-02	Rev. 1	Elaboration and review of regulatory standards and guides
PP-UPP-001	Rev. 0	Elaboration of Work Plan and Budget
PP-GSRN-001	Rev. 0	Record, assessment and event communication to INES.
PP-GSRN-002	Rev. 0	Issue and control of regulatory requirements fulfilment for Class 1 installations.
IT-GSRN-001	Rev. 0	Dose assessment.
IT-SGRN-001	Rev. 0	Communication of relevant events in Argentine nuclear research reactors.
PP-SGRN-004	Rev. 0	Relevant event management in nuclear reactors.
PP-SGRN-006	Rev. 0	Inspections and technical audits to research reactors and critical assemblies in operation.
PP-SGRN-007	Rev. 0	Regulatory tasks related to annual exercises of application of emergency plans in research reactors and critical assemblies.
PP-SGRN-009	Rev. 0	Routine evaluations of research reactors and critical assemblies.
PP-SGRN-011	Rev. 0	Non routinely evaluations in research reactors.
In preparation	.	Nuclear reactors licensing.
In preparation		Personnel training and re-training control for research reactors and critical assemblies.
PR-002	Under Rev.	Radiological emergency intervention. SIER system.

Loss of Power of a 220 VAC Safety Bus Bar at Embalse NPP

Perez, S.S.

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París, Francia, 11 al 14 de octubre de 2005

LOSS OF POWER OF A 220 VAC SAFETY BUS BAR AT EMBALSE NPP

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1- CHARACTERISTICS OF THE PLANT

Embalse Nuclear Power Plant is located at the Cordoba Province, 5 kilometres southern the town of Embalse. It is equipped with one 600 MWe CANDU reactor.

At the time of the event, which occurred on 11 October 2004, the plant was operating normally at full power.

The Embalse NPP electrical power system buses are classified, in reliability decreasing order, into the following four different levels: 1) Class I: Uninterrupted direct current supplies for safety related and other essential loads; 2) Class II: Uninterrupted alternating current supplies for safety related and other essential loads; 3) Class III: Alternating current supplies to essential auxiliaries which can tolerate the short interruption required until to start up and load the on-site standby generators and 4) Class IV: Normal alternating current supplies to auxiliaries which can tolerate long duration interruptions without affecting personnel or equipment safety (Class IV is the normal source of power to Class III system).

In particular, the Class II 220 VAC uninterrupted monophasic safety bus bars are composed by three 220 VAC subsystems powered by the following two sources of power: Class I uninterrupted safety bus bars (5551 BUA, BUB and BUC bars) and Class III secondary distribution 5433-MCC5 and MCC6. The Class II 220 VAC bus bars are normally powered from Class I bus bars through static monophasic inverters. Beside, in a static monophasic out of service case, Class II bus bar affected could be powered from Class III bus bars via a transformer stabilizer. Class II bus bars supplies power to both control computers and the AC instrumentation and control devices.

When 220 VAC (Class II) 5542-BUC-monophasic-inverter failure was coincident with a subsequent independent failure to transfer to the Class III through static commuter 5542-SVR-C, a loss of the 220 VAC uninterrupted monophasic safety bus bar (Class II) event was caused

2- EXECUTIVE SUMMARY

A monophasic inverter failure followed by the corresponding static commuter failure caused a 220 VAC uninterrupted monophasic safety bus bar (Class II) loss in the Embalse Nuclear Power Plant (CNE). The event occurred when the plant was operating normally at full power on October 10th 2004.

The event consequences were the loss of both, the turbine control system and the turbine steam by-pass. Besides, one (out of three) logic channel of some safety systems was tripped. However, the safety systems were not demanded to actuate. The back-up computer DCCY and five control room monitors were also lost.

The event is recurrent since it had happened on 31-01-00.

After a meticulous analysis (See Report CNE, Dip-10/00), corrective actions were taken.

Additionally to the recurrent event, during the shutdown progress, an independent failure consisting in an anomaly operation of an absorbent rod CA1, took place. The bar dropped more than it foreseen by the design. This bar belongs to the regulating system (not to the shutoff rods safety system).

3- EVENT DESCRIPTION

Failure in the monophasic inverter 5542-BUC followed by a delay of 4 seconds in energy transfer to bus bar 5542 BUC (220 Volt Class II). As a consequence of the double failure, what followed was electrical supply loss to Electric Hydraulic Control (EHC) system of the turbine causing turbine trip due to speed turbine measure and loss of the condenser steam discharge valves (CSDV) (blockage).

The failure of the single-phase inverter 5542 BUC 220 VCA and the subsequent failure of the static commuter 5542-SVR-C in the transference to the grid (with a delay of 4 seconds) caused that the bar 5542-BUC, was transitorily without energy, resulting as well in a lack of power supply to some relevant components

This caused an automatic power reduction through both reactor power set back and step back till 60 % full power (FP) due to turbine trip and then set back (0,5 % / sec.) till 15 % FP due to high steam pressure. The turbine trip caused steam pressure increase, opening the four steam discharge valves (ASDV) and three main steam safety valves (MSSV) atmosphere steam discharge valves (during 4 seconds)

Although the bar is classified as uninterrupted, losing it is essentially related with plant availability more than to safety concern. The safety concerns are the fact that a Generic Transient occurs demanding plant have to go to a safe shutdown state and a redundancy is lost for very short time in the DUAL COMPUTER CONTROL system, increasing slightly the frequency of the Initiating Event “Loss of Both Computers”. The add of the two facts is insignificant from PSA point of view.

The event 5542-BUC bar lack of electric power led to:

Trip (safe fail) of the logic channels F and J from Shutdown Systems 1 and 2, logic channel M of ECCS, logic channel Q from containment, loss of DCCY Computer (the back-up one) and lost of all measurements in process systems fed up from 5542–PL 568 (NSP side) and 5542-PL 654 (BOP side)

An independent failure occurred during the power reduction process: An absorber rod (CA1) went into the core until 75 % of the total length while the other component of the group was introduced until 45 %

Once restored the power at the 220-volt bar and achieved standard values of the most important parameters, reactor power was manually increased until 55 % full power to avoid the Xe 135 poisoning.

After 11 hours of the beginning of the event, the plant was synchronized to the interconnected system, reaching 100 % F.P. Bus bar BUC (class II) was connected directly to the grid (class IV) until the causes of inverter and commutator failures could be established.

4- REFERENCES TO PREVIOUS EVENTS (RECURRENCE)

The mentioned failures have antecedents in the existence of precursors between years 2000 and 2002. On January 31, 2000, a similar event happened with a similar evolution.

In addition during the shutdown process another independent failure consisting in an anomaly in the operation of absorbent rod CA1 took place (independent failure).

5- REGULATORY ACTIONS

During the investigation of causes, which produce the failure of 2 printed circuit board (PCB) of pulse modulation (considered the origin of the static commutator failure), the plant continued

operating without redundancy in the electrical system (connected directly to Class IV). This practice is not allowed **for longer than 8 hours** (Policies and Principles Manual).

Regulatory Authority received the evaluation carried out by plant experts, showing that the required level of safety has been preserved.

At CNE N.P.P, the following evaluations were carried out:

- A probabilistic assessment to verify the influence of the increasing of the frequencies of both the Generic Transient and the Loss of Both Computers initiating events, as well as the loss of one computer as redundancy as support system.

After that, the electrical supply to the only one safety related system (Liquid Control Zones) was changed to a secured bus bar. Therefore, the recurrence did not affect this system during 2004 event.

- In 2004, through a new evaluation of systems reliability, CNE experts verified that, in the existing scenario, all the systems connected to the bar would have “fail safe”.

An evaluation was done showing the low impact in safety for inverters out of order.

A decision is pending to define if Policies & Principles will be changed to allow longer time for inverters out of order.

6- SAFETY ASSESSMENT

1- DIRECT CAUSES

The direct cause of the event was the lack of power in the bar of 220 V Class II channel C (5542 BUC) as a result of the failure of 220 single-phase inverter BUC of 5542 V and the simultaneous failure of the on time transference to the supply from the grid caused by the failure of the static commutator 5542-SVR-C.

2- ROOT CAUSES

The root causes of the inverter failure could not be determined. Though an electronic 24 V source printed circuit board (PCB) CS 021 could cause a false contact at its point of connection.

The possible root cause of the failure of the static commutator could be originated in the failure of a pulse modulation PCB CS 204. To confirm this presumption, it is necessary to have a "test bank" that is not available at the CNE.

The cause of incorrect operation of absorbent rod CA1 is unknown yet.

3- CORRECTIVE ACTIONS

3.1 Performed corrective actions:

Repair of the inverter 5542 BUC and the static commutator 5542-SVR-C (by means of the execution of ID 94441).

Replacement of PCB CS 021. The new one is part of a new design group.

3.2 Pending corrective actions

Feasibility is being analyzed to install a system of energy supply to assure independence of the electro-hydraulic Control Panel of the turbine (ID 94632).·

The procedures of purchase of an inverter were reactivated in order to replace the failed 5542-BUC inverter (ID 83399).·

The deficiency detected in absorbent rod CA1 is being evaluated (ID 94502). During the time that it requires to execute this ID it was decided to suspend the monthly test of partial drop of this rod.

7- CONCLUDING REMARKS

The event is recurrent since an antecedent of similar failure happened at 31-01-00. At that time a meticulous analysis (to see Report CNE, Dip-10/00) was carried out.

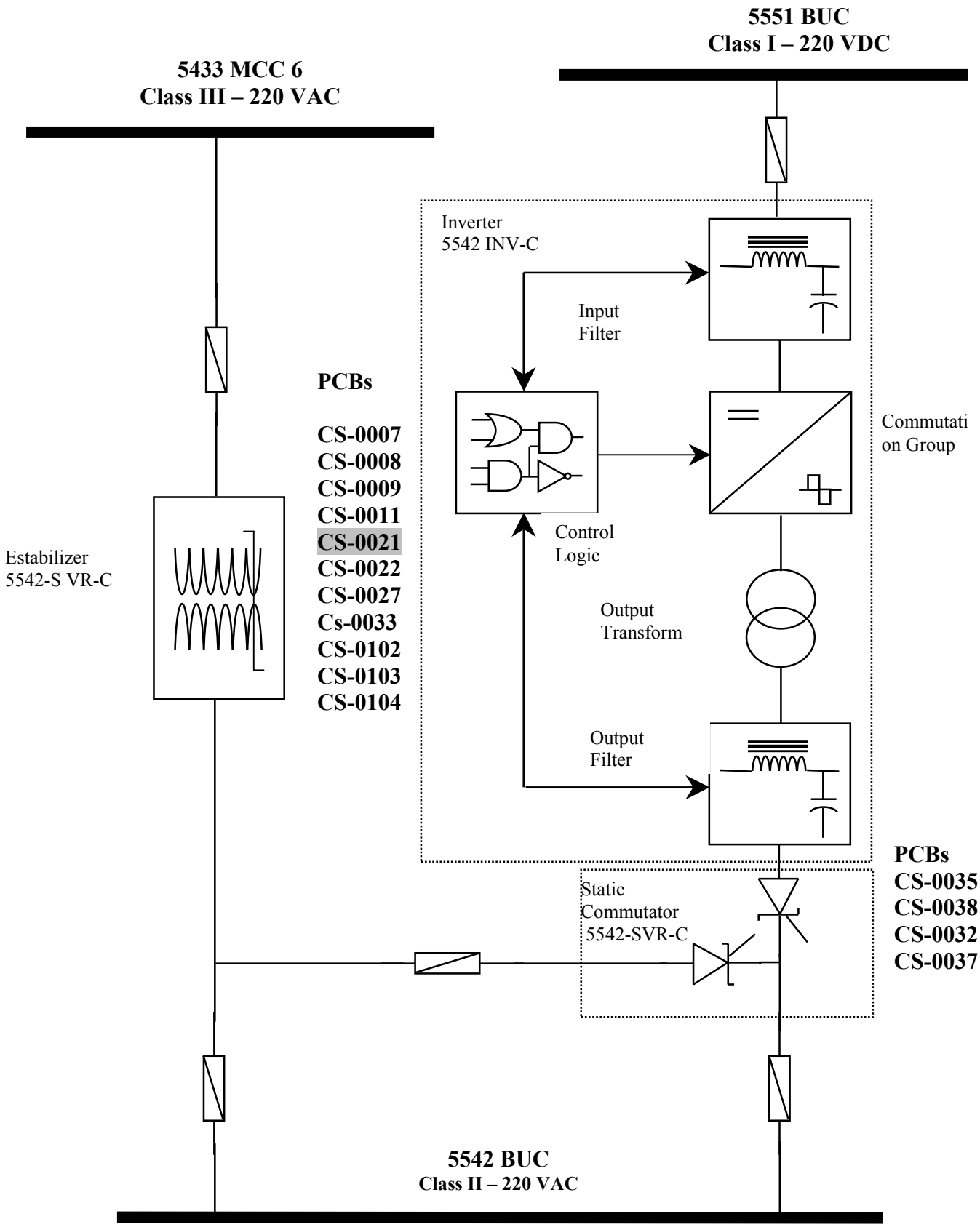
During the time in which the inverter stayed working in emptiness to check its performance (3 days), the plant continued operating connected to the grid without redundancy (from Class IV through Class III). It is important to mention that this practice is not allowed by Manual of Policies and Principles of Operation of the installation (point 5,15), but after the 2000 “Addenda” to Policies and Principles of Operation” it was verified (by reliability analysis) that it is possible to maintained the plant operating for more than 8 hours without an increase in risk

During the three days that the inverter was being checked in emptiness, a failure of printed circuit board (PCB) CS021 (part of the inverter) happened, attributed to a lack of continuity in a cold weld of the L2 resistance took place. This PCB was replaced (by one of a new generation) correcting the failure and the test continued.

The failure of absorbent bar CA1 (ID 94502) lead to suspend its routine test of partial drop until the cause of this failure can be determined and corrected. The mentioned test must be done with a monthly frequency, but a decreasing in the availability of one out of for bar of this system (regulation) does not influence in the plant safety in a significant way.

The no compliment of the established in “Policies and Principles of Operation” document occurred. However, the information related to the low impact of the event in plant safety had been taken into account.

ANNEX I


5551 BUC
Class I – 220 VDC
5433 MCC 6
Class III – 220 VAC
Inverter
5542 INV-C
Input
Filter
PCBs
CS-0007
CS-0008
CS-0009
CS-0011
CS-0021
CS-0022
CS-0027
Cs-0033
CS-0102
CS-0103
CS-0104
Commutation Group
Control
Logic
Estabilizer
5542-S VR-C
Output
Transform
Output
Filter
PCBs
CS-0035
CS-0038
CS-0032
CS-0037
Static
Commutator
5542-SVR-C
5542 BUC
Class II – 220 VAC

The Approach of the Nuclear Regulatory Authority of Argentina towards Nuclear Security

Racana, R.O.; Clein, D.A.; Rodríguez, C.E.; Nollmann; C.E.;
Tellería, D.M. and Fernández Moreno, S.

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THE APPROACH OF THE NUCLEAR REGULATORY AUTHORITY OF ARGENTINA TOWARDS NUCLEAR SECURITY

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ABSTRACT

The possibility that radioactive sources, nuclear and other radioactive materials could be intentionally used to harm and terrify people is not new. However, September 11, 2001 and other recent evens augment its likelihood. The increasing problem of orphan sources, the cases of illicit trafficking of radioactive materials and the attempts by sub-state actors to acquire such materials, have led to the adoption of concrete actions to strengthen global nuclear security, in particular since the mid 90's. The experience gained along these years to address these problems has been reflected in the conclusions of the preceding international conferences convened by the IAEA and other organizations and in several actions adopted at international and national levels.

We are of the view that global nuclear security strongly depends on the existence in each country of robust regulatory infrastructures in radiation and nuclear safety, physical protection, emergency preparedness and response and national safeguards that work together with other relevant organizations through a comprehensive and systemic approach, based on well-defined procedures and standards to ensure adequate and timely response to potential malicious acts involving nuclear and other radioactive materials. This presentation describes the approach followed and the actions taken by the Nuclear Regulatory Authority of Argentina to enhance its capability to prevent and respond to such malicious acts, the experience gained to date and the future steps under consideration to maintain and improve such capability. It also provides a broader argentine perspective on nuclear security and non-proliferation.

INTRODUCTION

One of the findings of the "International Conference of National Regulatory Authorities with Competence in the Safety of Radioactive Sources and the Security of Radioactive Materials" held in Buenos Aires in December 2000, was the need to differentiate clearly those situations in which people are unintentionally exposed to radiation from those where there is a criminal intent to irradiate other individuals, to frighten or produce disruption to the people. At the International Conference held in Stockholm in 2001, we noted, "*The existence of competent regulatory authorities that set up adequate control measures is an essential component to lower the probability of occurrence of illicit uses of nuclear materials"*. Later on, in March 2003, at the International Conference on Security of Radioactive Sources held in Vienna, we said that "*the novelty of the new scenario is that it includes the malicious intention of those who could take possession of radioactive sources and other radioactive materials with the suicidal attitude to reach their goals, increasing the probability of occurrence and challenging the control system."* We consider that these findings remain valid today.

DESCRIPTION OF THE PROBLEM

We are facing a multifaceted problem characterized by the coexistence and combination of different possible scenarios. Although the threat of malicious acts involving radioactive materials is global, there is a need for each country to assess and establish its own credible threat scenarios to adopt the measures that better address them, aiming at achieving a reasonable level of protection without posing an undue burden to the beneficial uses of radioactive sources.

To identify the regulatory approach to possible malicious acts, consideration should be given to the following facts:

- There are still a significant number of orphan sources that could not only be found by innocent people or be subject to inadvertent movements, but also could be taken by people with a malicious intent.
- The existence of illicit trafficking of radioactive and nuclear materials and the possibility that these materials be used with a malicious intent.
- The attempts to acquire such materials or to sabotage a relevant installation with the aim of harming or terrify people or to cause disruption.

We consider that there is not a *"one fits all formula"* to address these realities, but the existence of robust regulatory infrastructures and concerted actions taken at the international level certainly reduce the likelihood of malicious acts involving radioactive materials. Measures to respond to such acts, in particular those associated with nuclear terrorism, should be commensurate to those credible threats that each country has established. A step-wise approach is advisable, in particular, to recognize that is not possible to protect radioactive materials from all possible threat scenarios associated with potential malevolent intents. The variables involve are just infinite, thus the addition of security measures to the various applications of radioactive materials should be based on the definition of credible situations, taking into account the risks associated with the sources. In this context, emphasis should be given to the prevention and response of such events for high-level radioactive sources.

ACHIEVEMENTS AND LESSONS LEARNED

Since the 50s, Argentina has established a control infrastructure based on the knowledge of the risk associated with the deleterious effects of ionizing radiation and on the international philosophy for their assessment and limitation. Registration, licensing and control of radioactive sources from "the cradle to the grave" have been essential features of our regulatory system. In this context, safety and security have been always closely related concepts.

As the nuclear regulatory authority with federal competence in radiation protection, nuclear safety, safeguards and physical protection, the Nuclear Regulatory Authority of Argentina (ARN) has established standards and regulatory requirements. It is also responsible for issuing licenses and permits to any activity involving radioactive materials and for controlling and verifying that these activities are performed in full compliance with ARN standards.

Nuclear Security

With regard to nuclear security, the ARN has adopted a systemic approach at a national level based on the extrapolation of some physical protection criteria to the use, storage and transportation of certain radioactive materials. It has also actively participated in the international efforts towards increasing nuclear security worldwide. On the other hand, ARN considers it important the IAEA promotion of the use of knowledge management techniques to develop process flows, map safety knowledge and to encourage knowledge sharing and the establishment of regional nuclear and radiation safety networks to preserve and strengthen existing knowledge and expertise in these fields. Prominent examples are the Asian Nuclear Safety Network established in the frame of the Agency's Programme on the Safety of Nuclear Installations in South East Asia, Pacific and Far East Countries, and the Ibero-American Radiation Safety Network in the frame of the Ibero-American Forum of Nuclear Regulators. Results to date are most encouraging and suggest that this pioneer work should be extended to other regions and eventually to a global safety and security network.

Moreover, since the very beginning, Argentina has been involved in the preparation and process of approval of the Code of Conduct on the technological and physical security of radioactive sources and it is in full compliance with the criteria established in the Code. This has also been made clear when Argentina ratified its commitment to the Director General of the International Atomic Energy Agency to support and to follow such guidelines.

Import and export of radioactive materials

The Nuclear Regulatory Authority is empowered to ensure that the import and export of radioactive materials are made in agreement with the criteria established in the Code of Conduct, and in particular, with the guidelines for the import and export of radioactive sources. These guidelines will enter into force towards the end of 2005 beginning of 2006, in spite of which Argentina has already implemented them fully.

“Registered” radioactive sources and materials

“Registered” radioactive sources and materials are under regulatory control; thus their illicit uses would be prevented by this system or in case of a breach involving those materials, there is a response scheme to allowing the implementation of rapid remedial actions. On the other hand, despite the low probability of occurrence of illicit trafficking in the region, we can not exclude it. Therefore, it has been necessary to adopt measures of "security " in addition to those of existing radiation safety. In addition to this scenario, the possibility of malicious acts such as the robbery, theft or sabotage to facilities with high radioactive inventories, still exists and the same being true in the case of transport of radioactive materials with a significant activity within the national territory. A case under study by the ARN is the transport of Co^{60} as Argentina is one of the main producers of Co^{60} and normally makes transports that involve inventories in the order of 55 TBq ($1,5 \times 10^6$ Ci) of Co^{60}. The measures adopted are similar to those of physical protection for the transport of nuclear material. In the case of installation handling high radioactive inventories[1], the ARN has established additional security measures also based on its approach to physical protection.

Emergency Preparedness and Response

Originally, the ARN was the one who defined the requirements for the regulated in matters of prevention and preparation to face radiological and nuclear emergencies and advised the acting municipal, provincial and national official organizations. In addition, ARN always had its own “on call” specialized operative groups able to respond to emergency situations with radioactive sources, to act when the capacity of a facilities was surpassed or in emergencies that took place in the public thoroughfare (for example, in transport related accidents, etc.). In Argentina, an average of 10 annual interventions of minor emergencies have been taken care of in the last 40 years and the ARN participates directly accompanying the intervening security forces (Firemen, Police, Gendarmerie, Prefecture, etc.)

In the area of emergency preparedness at relevant nuclear facilities, the ARN not only established the requirements and advised the acting official organizations, but also controlled and participated actively in the emergency drills for more than 20 years. In 1997, when the Nuclear Act was enacted, ARN has been assigned with the function "to direct the emergency actions during nuclear emergencies in the off-site area". From then on, the ARN begins to organize, train, act and direct nuclear emergency drills, where the coordinated direction of all the civil organizations, security and armed forces is performed, all lead by a centralized command lead by an ARN operation head whenever the ionizing effects of radiation is concerned. For this new function the ARN follows the guidelines established in "Preparedness and Response for a Nuclear or Radiological Emergency" international requirements.[2]

Specific training courses

Among other works in progress, the training courses and awareness, seminars at national and regional levels are of outstanding importance, not only in the case of operators but also for the different control and security organizations participating in the approach. In this context, it is worthy to note that ARN has planned intensive training programs at national and regional level (i.e. trough the Security Commission of MERCOSUR). These specific training courses are delivered in cooperation with IAEA, INTERPOL, DOE, National Customs Agency, National Response Forces and Intelligence Community.

[1] Class I installations, according to the classification of the IAEA TECDOC/1344.
[2] The FAO, IAEA, ILO, OCHA, OECD/NEA, PAHO, WHO, GS-R-2 Requirements.

CONCLUSION

In sum, we can say that ARN is executing different activities in the fields of prevention; legislation, response, training and exchange of information to further enhance physical protection of nuclear material and the security of other radioactive materials. We have come to the conclusion that the most effective approach to nuclear security towards the prevention, early detection and response to illicit or malicious acts involving radioactive materials are realized through the existence of a robust regulatory infrastructure in each country, a permanent exchange of information and contact between the nuclear regulatory body, customs authorities, intelligence services and security forces and the appropriate training of relevant staff. These are essential constituent parts of a systemic process that implies knowledge and assumption of responsibilities by all the participating institutions, working in a coordinated manner.

A BROADER ARGENTINE PERSPECTIVE ON NUCLEAR SECURITY AND NON-PROLIFERATION

Argentina, as a way to fight these new threats concentrated all its efforts to make the non proliferation regime more effective, more efficient and reliable. Last year, my country has chaired two export control regimes: the Wassenaar Arrangement and the MTCR. Argentina intends to contribute to strengthening the export control regimes while making good use of them. Moreover, currently Argentina participates in all of the existent multilateral disarmament and non-proliferation regimes.

As regards domestic legislation, the Argentine Republic has implemented a strict control upon the international transfers of certain materials, equipment, technology, technical assistance and services of nuclear and missilistic nature, as well as on chemical and biological substances and war materials, which may contribute to the production and dissemination of missiles, nuclear, chemical or bacteriological weapons.

Argentine regulations do not restrict the lawful trade, but introduce Non Proliferation international criteria to the national legislation.

In that sense, Argentina welcomed the adoption of Security Council Resolution 1540 (2004) on “weapons of mass destruction and non state actors” that affirms that “proliferation of nuclear, chemical and biological weapons, as well as their means of delivery, constitutes a threat to international peace and security”. In October 2004 Argentina submitted its first National Report to the 1540 Security Council Committee.

Besides, Argentina supports the changes to be introduced in the framework of the International Maritime Organization (IMO) to the “Convention for the Suppression of Unlawful Acts Against the Safety of Maritime Navigation” to include offense for transporting WMD and non proliferation offense that the original Convention did not cover.

Securing nuclear and radioactive materials that pose a threat to the international community through the Global Threat Reduction Initiative (GTRI) it is also a priority. Since the very beginning Argentina took part of the proposal, we have supported the effort to reduce world stockpiles of highly enriched uranium (HEU) employed in civilian nuclear programs. Our experience in this field has been extended to other countries’ consideration. Starting in October 2004 we have hold several meetings with the U.S. Department of Energy noting that there is an extensive list of specific technical activities linked to reactors conversions that could integrate the efforts of both countries in order to address the objectives of GTRI.

Tackling illegal trafficking trespassing our territory is another part of the same problem. With this aim Argentina is negotiating a Memorandum of Understanding with the United States to implement the “Megaports Initiative” to detect unauthorized transport of radioactive materials. Considering the important activity of the port of Buenos Aires this initiative reaches the political and technical elements to project the experience to the whole region.

“SYSTEMIC PROCESS” APPROACH

We are facing a complex problem that requires worldwide attention, which should be addressed by each specific country within the framework of an international system. To prevent and respond to possible malicious acts involving nuclear and other radioactive materials, the ARN is working intensely to develop a different types of drills base on a "systemic process" approach.

MANAGEMENT PRINCIPLES

ARN is introducing some principles to handle this new challenge, the purposes of which are to design and implement a quality management system to achieve the security goals. The greatest value would be obtained when the design became tailored to the quality management standards and guidelines known as the ISO 9000 family. They are very useful and practical for established quality systems in general and therefore would be applicable to this particular case. These documents introduce eight quality management principles, which are derived from the experience of the international experts who participate in ISO Technical Committee ISO/TC 176, *Quality management and quality assurance*.

These principles will be used by the ARN to guide and control the security design towards improved efficiency and performance, because a security system must be a quality management system.

The use of these principles requires:

1. Understanding of behaviors and attitudes of the people direct and indirect involve in the process. Starting from this point, present and future security projects must be designed and their concept communicated the main idea being that the projects should satisfy the expectations. The projects should also ensure a balance between the relevant safety and security requirements. To reach these objectives, it would be essential to establish:
    a. unity of purpose,
    b. a clear vision of the problem,
    c. common objectives and
    d. shared values.
2. People motivated, with trust and no fears in the process, with the required vision, adequate resources and a good training, They should understand the methodology and move towards the goals and objectives desired by the ARN. If only the scientific approach to solve the problem would be considered, for example, ignoring the people, the approach may produce negatives reactions.
3. Managing by process the organization, the responsibilities, the authorities, the procedures, the standards, the resources, and the activities. That is to say: there should be a management by process rather than a classic division of responsibilities by function. The process requires a manager, whose main responsibility should be to define the design basic threat and also the procedures and the standards. The participants must acts coordinated by procedures previously established a priori rather than by giving orders a posteriori through a hierarchy structure. The advantage of this approach is that the response to possible malicious acts be prompt and systematic. The process would include not only aspects of security, but also the response to emergencies of great magnitude in terms of radiological consequences, be they caused by terrorist attacks or, very low probability severe accidents. Presently the functions are being distributed according to the specific capacities of each organization involved (civil, local and national police, intelligent system, customs, security and armed forces) all of them forced by their own constituent laws to act in cases of disasters to protect the population and in the framework, knowledge and procedures established by the manager of the process.
4. Design the processes as a global system and apply the principle of continual improvement. This approach would facilitate understanding of the interdependencies between the activities, tasks and processes, not only by the participants but also by the population as a whole, and to achieve the goals in the most effective way. The decisions must be taken through reference to factual

records, which imply accurate and reliable information, for this reason the methodology to design the processes should be based on experimental approach.

In summary: in order to design and implement a working security system it is necessary to:

- communicate a clear vision,
- motivate a global participation,
- organize the project like a network of processes, and
- build the model in an experimental way.

THE GENERAL STRATEGY FOR A PLAN OF ACTIVITIES TO COMBAT NUCLEAR TERRORISM: THE SECURITY PROCESS.

In order to obtain the objectives, a comprehensive management strategy should be implemented based on the documents in the ISO 9000 family.

This part describes the methodology and actions that are required in each phase, with particular emphasis in the "early alert" and the "response" phases. For these phases, the systemic process includes:

- all national organizations that should be involved in the assessment and establishment of the threats and the timely exchange of information and
- the ones that may be involved in emergency situations originating from significant damages in relevant nuclear facilities (e.g., NPPs. and other installations holding a high inventory of radioactive materials).

The methodology would require several stages to get the objective. The first is to establish the step-by-step process and sub processes required. In order to raise levels of quality, safety, reliability and efficiency, the methodology should be based in a combination of process. Processes require resources and must be managed to achieve the desired output. A sub process' output is the next sub process' input. The final product is often the result of a system of processes.

In the problem under consideration, the process is a "Security Process". The security process is not an independent process by itself but rather it is part of a network of other process, mainly the "Safety Process.

The second stage of the process design would require a documentation plan. This plan should include:

- training,
- communication,
- standards,
- procedures and
- assessment.

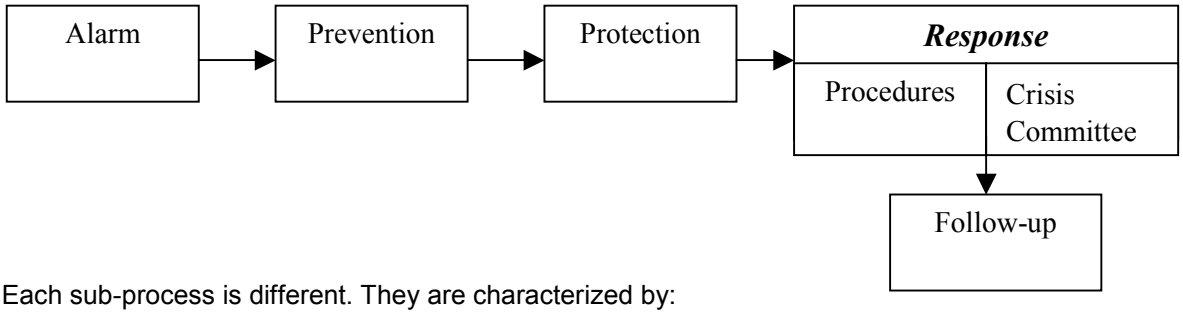
With all of these elements a Plan should be formulated in order to implement the processes.

A thorough review of the ARN has indicated that all the procedures used previously were applicable to the above described new quality management system. However, additional procedures had to be included in order to give a systematic approach to the problem.

THE BASIC DESIGN OF THE NEW APPROACH

The basic design of the new approach comprises five sub processes:

i) Early alert / alarm,
ii) Prevention;
iii) Protection;
iv) Response and
v) Follow-up actions.



Each sub-process is different. They are characterized by:

- the quality and type of those who participate in the sub-process,
- by the number and type of organization to which they belong
- the participants' knowledge, and
- the goals each sub-process must reach.

The manager of the security process should define:

- the design basis threats, and
- the standards and guidelines.

Moreover, the manager should think in terms of quality management system, management responsibility, resources, measurement, improvement, procedures and trainer of all participants.

THE SUB-PROCESSES

Alarm: In this first stage, the competent intelligence agencies should be strongly involved (In the specific case of Argentina, the national intelligence organization is properly advised on technical nuclear matters by the ARN). The intelligence agencies should have the mission to detect the potential threats to divert nuclear and other radioactive material and to attack relevant facilities or transportation.

Prevention: In this second stage, ARN has also an important role and the customs, national security forces, national and local police and other law enforcement agencies should be strongly involved. They should prevent and avoid theft, sabotage and other malevolent actions, by means of the relevant control system spread under the territory under their jurisdiction.

Protection: In the stage of protection, relevant national forces should be assigned with specific responsibilities. In the specific Argentine case, while the National Gendarmerie has particular responsibilities on protection of nuclear installations, the ARN begins to have active functions through their regulatory power and in the preparation of its own capacity to face possible radiological consequences of attacks or severe accidents. The forces are previously defined and the locations and magnitude depend of the facilities and transports of materials that must be protected.

Response: The first phase of the response stage is defined by the application of a priori established procedures. This takes place during the first hours of a serious security breach. While differences between the emergency created by a security breach and those resulting from an accident must be recognized, however there is at this stage where a connection exists between the radiological emergencies that may result from a failure in safety and those created by a security situation. Whenever it is not possible to make detailed evaluation previously defined and tried countermeasures must be applied. Later, when detailed evaluations need to be made a Crisis Committee, to be established a priori, should play the most important role in the subsequent decision-making process. In the specific case of Argentina, the ARN is designing and ad hoc emergency drill. The design will be ready in few weeks by the beginning

of April. The purpose is being able to organize a definitive drill towards the last quarter of 2005. The ARN is the organization who leads the activities that promote the improvement of the response capacity of the Argentine State, not only in the local area in the first 24 hours (where the first effort is put), but also on a regional and national scale, during the later stages, that may reach distances of several hundreds of kilometers and periods of weeks, months or years.

Follow-up: The main objective of this phase is carrying out:

- Studies and assessment of the work that have done introducing the modifications to be applied in the next process, and
- the verification of compliance of the process with relevant measures and requirements.

FUTURE STEPS

The Nuclear Regulatory Authority is considering means to further improve its capability to prevent and respond to malicious acts involving radioactive materials. Future actions include the strengthening of the systemic approach at national level by consolidating the activities foreseen in each of the above-mentioned phases, to increase its participation in the international efforts underway, particularly within the framework of the IAEA activities and the “Global Threat Reduction Initiative” and other relevant multilateral actions.

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El responsable por la seguridad radiológica en la industria. Entre el ambiente laboral y la tecnología de hoy

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EL RESPONSABLE POR LA SEGURIDAD RADIOLÓGICA EN LA INDUSTRIA. ENTRE EL AMBIENTE LABORAL Y LA TECNOLOGÍA DE HOY

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1-RESUMEN

Dentro de las aplicaciones industriales de las fuentes radiactivas selladas, existen dos ramas claramente definidas para los cuales estos materiales son utilizados: fuentes radiactivas utilizadas en equipos fijos y en equipos móviles. Estos dispositivos se utilizan en un sinnúmero de aplicaciones y cada vez con sistemas tecnológicos más avanzados. Esto requiere una mejora permanente dentro de la capacitación del responsable, y a su vez introduce un cambio obligatorio a otros grupos de intervención durante la vida útil del dispositivo. Los riesgos de accidente, a los que se encuentran asociados el uso de estos equipos, si bien su tasa de ocurrencia es baja, son errores durante la operación, uso indebido, mantenimiento realizado por personal sin conocimiento, fallas de tipo humano o descuidos e incidentes ocurridos durante el transporte. Todos estos "riesgos" están rodeados de distintos grupos de factores que influyen durante el empleo seguro del material radiactivo dentro de la instalación de una manera u otra. Entonces se destaca la importancia de implantar aspectos de "Cultura de la Seguridad", evaluaciones de aptitud del responsable por la seguridad radiológica, introducción de mejoras en aspectos de calidad y comunicación de riesgos, acorde a lo que indica la normativa de la República Argentina y los requerimientos internacionales. Desde el punto de vista de la concepción de las fuentes selladas, estas son intrínsecamente seguras si son utilizadas en forma correcta, dado que cumplen con los ensayos necesarios (ISO 2919), y a su vez, se encuentran dentro de un blindaje adecuado provisto por el fabricante del equipo, el cual responde un diseño seguro (ISO 7205), lo cual en condiciones normales de operación brinda la protección necesaria, y los valores de tasa de dosis emergentes de los mismos, permiten establecer límites aceptables para el operador desde el punto de vista de la protección radiológica.

Si bien la mayoría de este equipamiento es de simple operación, en la práctica, descuidos por parte del usuario autorizado, falta de recaudos de seguridad o falta de "actitud", son los causantes de incidentes o accidentes en los cuales el factor humano es quien determina la magnitud del mismo.

En el presente trabajo haremos referencia a todos los conceptos que rodean la elección del responsable del material radiactivo, su capacitación, los factores que intervienen durante la operación, las obligaciones de la instalación, situaciones de inseguridad y los cambios o mejoras que introduce la tecnología.

2-DESARROLLO

En los últimos años incidentes o accidentes han sido reportados a los sistemas de información internacional para su divulgación y conocimiento (Sistema **NEWS** – **N**uclear **E**vents **W**eb Based **S**ystem)

De la información recibida se detecta en general, situaciones derivadas del desconocimiento o desinformación o de la falta de "percepción del riesgo" por parte del usuario de material radiactivo.

Este aspecto, que resulta no ser menor al momento de la toma de la decisión correcta, una vez detectado el incidente o accidente radiológico, expone la necesidad del desarrollo de un plan de contingencias para remediar de manera rápida y eficiente las consecuencias del hecho. Esto quiere decir que una pobre cultura en lo que se refiera a la percepción del riesgo se traducirá en una complicación mayor aún al momento de intentar resolver la emergencia sin la preparación y los cuidados necesarios.

Figura 1. Ejemplo de equipos para medición de humedad y densidad dañados por descuido.

Desde el punto de vista de la seguridad radiológica, el uso de fuentes radiactivas selladas utilizadas durante una práctica normal con material radiactivo produce dosis muy bajas o planificadas, pero durante una práctica anormal o errónea se producen dosis no planificadas o injustificadas.

Los accidentes radiológicos por lo general se producen durante:

- ✓ La operación normal.
- ✓ En tareas de mantenimiento.
- ✓ En el transporte.

Estos accidentes son generados por:

- ✓ Fallas humanas.
- ✓ Fallas de procedimiento o sistemas.
- ✓ Una mezcla de ambas.

Esto quiere decir que para evitar un accidente o incidente de tipo radiológico, debemos tener en cuenta una serie de factores, los cuales podemos dividir en tres grandes grupos:

Factores del grupo I, Factores del Grupo II y Factores del grupo III.

3-FACTORES DEL GRUPO I

Este grupo de factores está dirigido a la instalación y a la elección del responsable, en general y a su capacitación:

- La instalación debe elegir al personal que posea la mayor “capacidad” para el manejo de materiales peligrosos, en este caso, material radiactivo.

- La instalación debe cumplir con todos los requisitos en materia de seguridad radiológica, según el riesgo que represente su instalación para los trabajadores y los miembros del público.

- Los requisitos de capacitación para el manejo seguro del material radiactivo deben de ser satisfechos en el 100% de sus requerimientos.

- La formación general y específica es el punto más importante. No es aceptable una sin la otra, como así también, el re-entrenamiento y la actualización.

- Debe emplearse una fuerte corriente de formación, íntimamente ligada a la cultura de la seguridad y al código de conducta en el uso de fuentes radiactivas, como así también la adecuada protección física de estos materiales.

3.1-La elección del responsable.

El comienzo del proceso de elección del individuo que actuará como responsable por la seguridad radiológica, se inicia en la propia empresa, que utilizará el material radiactivo como control de calidad o control de proceso, o en algún uso particular, como los empleados en tareas relacionadas a trabajos en pozos de petróleo.

En este paso inicial, tiene suma importancia la posición que ocupa actualmente la persona elegida dentro de la instalación, esto quiere decir, debe tener vínculo con las tareas de mantenimiento, higiene y seguridad, instrumentación, etc., por citar algunas de las ramas que intervienen durante la operación de la instalación. Esto significa una decisión crítica desde el punto de vista de la seguridad, dado que se estaría incurriendo en una falla en la elección del responsable por la seguridad radiológica por parte de la institución. El resultado es capacitar a la persona incorrecta.

Esto involucra de manera mucho más firme a la persona elegida, que en el caso de cumplir alguna función de tipo administrativa, ya que esto conlleva a desconocer algunas tareas de ingeniería u operación, dentro de la propia instalación.

En este punto la dependencia correspondiente (recursos humanos, capacitación u otros) debería realizar una valoración del perfil de cada uno de los individuos que reúnen los requisitos generales, (formación integral, responsabilidades dentro de la instalación etc.), para así llegar a aquel que por razones particulares (liderazgo, criterio y actitud) deberá ser el elegido, sin olvidar los estudios psico-físicos, que adquieren en la actualidad una importancia mayor día a día.

Dentro de los parámetros psico-físicos realizados en la industria de hoy, aparecen algunos indicadores como ser:

- ✓ Carga física laboral.
- ✓ Factor humano.
- ✓ Siniestralidad.

Estos parámetros son importantes en todo tipo de industrias, en general. Pero son mucho más importantes, si entendemos el mismo concepto aplicado al uso de sustancias peligrosas (material radiactivo, gases venenosos, líquidos inflamables, explosivos, etc.), donde los riesgos aumentan en forma considerable tanto para los trabajadores de la instalación como para los

miembros del público, en forma gradual, de acuerdo a la peligrosidad asociada a cada proceso industrial.

Durante los últimos años han ocurrido accidentes e incidentes de tipo radiológico en los cuales se han visto una serie de fallas y errores humanos, los cuales solos o en conjunto han generado exposiciones a la radiación injustificadas.

A continuación se realiza un comparativo sobre la probabilidad de ocurrencia de accidentes por año durante el empleo de material radiactivo en la industria en general. (Fig. 2)

	Cantidad anual
Accidente con consecuencia importante	**Uno**
Accidentes con consecuencias menores	**Algunos**
Accidentes con daños a la propiedad o al medioambiente	**Muy pocos**
Incidentes que no involucran lesiones o daño	**Una decena**
Situaciones de inseguridad o actitudes inseguras	**Varias decenas**

Figura 2. Tabla de comparación sobre accidentes y actos inseguros.

3.2-La capacitación.

El nivel de capacitación deseada por el solicitante de una Licencia de Operación para el responsable del uso de material radiactivo debe ser el más elevado que pueda conseguir.

Para ello se debe dividir la capacitación en diferentes etapas:

- Entrenamiento inicial.
- Entrenamiento general.
- Entrenamiento específico.
- Entrenamiento de actualización.

Estos niveles deben ser alcanzables en un determinado período de tiempo, que permitirá a la empresa analizar el grado de desempeño del personal seleccionado y la evaluación de distintas aptitudes, que calificaremos como indicadores de “performance”.

Entre ellas resaltaremos:

- Grado de responsabilidad.
- Manejo de personal.
- Habilidad para el análisis de situaciones de complejidad.
- Habilidad en la comunicación hacia el personal.

Los programas de capacitación deben de incluir aspectos como:

- Revisión de conceptos sobre radioprotección y cultura de seguridad.
- Adecuación de procedimientos de emergencia y estrategias dentro de la instalación.
- Mejoras de tecnología.
- Resultados de controles o auditorías.

- Realimentación de experiencias en otras instalaciones.
- Lecciones aprendidas de accidentes o incidentes radiológicos.
- Mejoras de calidad.
- Adecuación a nuevas normativas o reglamentos de aplicación.

3.3-Percepción del riesgo.

En el uso de material radiactivo es necesaria una cualidad adicional en el responsable por la seguridad radiológica y es donde aparece una condición natural de los individuos, que es la percepción del riesgo.

No todos la interpretan de la misma forma. La actitud de las personas **esta relacionada directamente con la percepción del riesgo al que están expuestas.** No alcanza con saber que existe un riesgo. Tenemos que conocer el problema en toda su magnitud, como se relaciona con la tarea habitual, la necesidad de implementar medidas para la prevención y el análisis de los procedimientos utilizados. El nivel de percepción depende fundamentalmente del grado de conocimiento y de la capacitación adquirida. Personas menos capacitadas o instruidas tendrán un grado de percepción menor, lo que traducido, generará un riesgo potencial elevado. Este individuo, será extremadamente propenso al accidente y para que ocurra el mismo, solo será cuestión de tiempo.

3.4-La inseguridad.

Es importante distinguir algunos síntomas que revelan una probable condición insegura en el responsable por la seguridad radiológica:

- ✓ Las actitudes del individuo.
- ✓ La conducta del individuo.
- ✓ El clima laboral.
- ✓ La presión laboral.

Estos factores normalmente contribuyen al deterioro de la seguridad radiológica de una instalación y serán tanto más evidentes cuanto más “detalles” o irregularidades se detecten durante las inspecciones regulatorias.

Esto quiere decir que los responsables por la seguridad radiológica deben ser personal seleccionado por sus cualidades. Deben ser operadores o usuarios cuya formación, capacitación y percepción del riesgo radiológico sea más elevada, y para ello la instalación deber realizar un juicio de valores, dado que la misma asume la responsabilidad total por la posesión del material radiactivo, lo cual debería ser una condición, sin la cual, no debería emplear materiales peligrosos, con el serio compromiso que representa ante la sociedad.

3.5-Aspectos de Calidad.

Dentro de este análisis no debemos olvidar la introducción de los conceptos de calidad y garantías de calidad.

En el mundo actual, las instalaciones industriales tienden hacia la calidad y la mejora continua.

El objetivo de un sistema de calidad es implementar el modelo adecuado y aplicable a las características de la instalación.

La calidad en protección radiológica es sinónimo de control. (Ej.: si se toman los recaudos necesarios se evitarán fallas, a menor cantidad de fallas, menor cantidad de accidentes o incidentes).

Para diseñar un buen sistema de calidad, entonces, debe conocerse profundamente los principios y necesidades de la instalación.

En este aspecto el responsable por la seguridad radiológica de una instalación se convierte en un factor de **“calidad”**, si realiza lo siguiente:

- ✓ Establece procedimientos que incluyan aspectos de cultura de la seguridad y los mantiene vigentes.
- ✓ Mantiene las condiciones de seguridad radiológica y física de las fuentes radiactivas a su cargo.
- ✓ Crea conciencia entre sus colaboradores.
- ✓ Tiene una elevada percepción del riesgo radiológico.
- ✓ Trabaja con seguridad y tiende a la propia mejora y la de sus colaboradores.
- ✓ Es constante en su tarea.
- ✓ La eficacia y la eficiencia son sus metas.

4-FACTORES DEL GRUPO II

Estos factores se dividen en fallas de evaluación de procedimientos, análisis de desempeño y errores o situaciones de inseguridad por parte del responsable por la seguridad radiológica. Entre ellas:

- Fallas de capacitación en protección radiológica.
- Ante todo pensar que los equipos que contienen material radiactivo “son seguros”. (exceso de confianza).
- Fallas en los enclavamientos de la instalación.
- Fallas de protección física adecuada, aplicada a las fuentes radiactivas o los equipos.
- Falla de señalización.
- Ausencia de monitor o detector de radiación adecuado.
- Aquellos que lo poseen, no realizan tareas de monitoreo durante la operación.
- Ante malfuncionamiento de un medidor nuclear con fuente radiactiva se realizan maniobras “ingeniosas” para hacerlo funcionar, aún a sabiendas de que probablemente esto generará una situación riesgosa.
- Muchas veces se trabaja “solo” (inseguridad o incompetencia).
- Ausencia de procedimientos escritos.
- En los casos que los poseen, estos procedimientos no son respetados.
- Falta de un programa de calidad.
- Falta de un programa de gestión de fuentes en desuso.
- Aspectos relacionados con la cultura de la seguridad.
- No se siguen los criterios en materia de seguridad radiológica.
- Fallas cometidas durante el transporte.

En virtud de ello se deben desarrollar e introducir mejoras permanentes en los conceptos tanto de los requisitos para obtener la licencia de la instalación correspondiente como en el permiso individual del responsable, reforzando lo referido a la actitud de los mismos, sus obligaciones y a los criterios a aplicar, desde el punto de vista de la seguridad radiológica durante todo el tiempo que utilice material radiactivo en la instalación.

5-FACTORES DEL GRUPO III

Este grupo reúne la última clase de factores que intervienen de alguna manera en el responsable o usuario del material radiactivo, y son consecuencia de la comprensión, la tecnología y la modernización:

- Minimizan los agentes externos que pueden producir incidentes (falta de atención).
- Condiciones del entorno (rutina).
- Manejo inadecuado del equipo a su cargo o desconocimiento.
- Falta de motivación.
- Las recomendaciones realizadas durante las inspecciones regulatorias no son observadas en el uso rutinario, solo cumplen con ellas como un **“requisito regulatorio”.**
- Falta de actitud en su tarea habitual.
- Falta de la aplicación de procedimientos durante una emergencia.
- Falta de notificación inmediata ante un suceso anormal a la Autoridad Reguladora.
- Falta de conocimiento de los procedimientos aplicables a su tarea.

- Falta de pautas claras durante el proceso de cambio o mejora de la institución.
- Negativa a efectuar su tarea de una manera distinta o diferente (costumbres).
- Falta de recursos suficientes para la tarea (personal, tiempo de dedicación, etc.).

5.1-La Tecnología.

El desarrollo de equipos modernos, hoy pone a la industria en un plano de actualidad tecnológica creciente, y los cambios que se produzcan deben de ser acompañados por un crecimiento, tan profundo, cuanto más profundo y complejo sea el equipamiento que contiene material radiactivo. (Fig. 3 y 4)

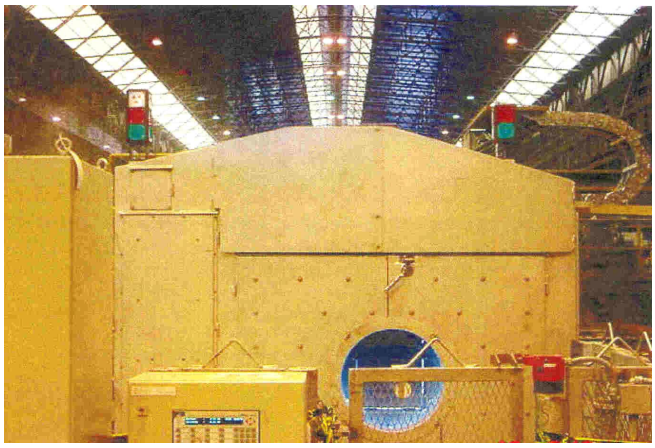

Figura 3. Ejemplo: Equipo medidor de parámetros estructurales de tubería de acero, conteniendo 9 fuentes radiactivas de Cs-137 (total 3,33 TBq)

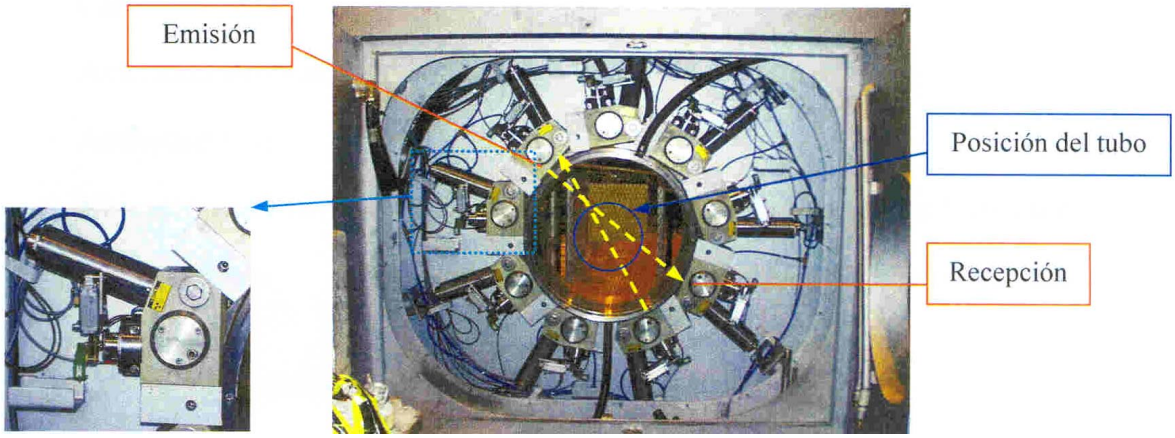

Figura 4. Detalle de la disposición de los 9 conjuntos fuente-detector.

Estos equipos utilizan actividades más que considerables, y poseen sistemas de diferente complejidad tecnológica. (apertura y cierre de obturadores (manual y automática) , indicadores luminosos y acústicos de la condición del equipo (fuente radiactiva en posición ON-OFF), trabajan a temperaturas sumamente elevadas (del orden de 900°C), poseen sistemas de enclavamiento para restringir el acceso al área de trabajo, protección física obligatoria, alarmas por malfuncionamiento de algún componente, sistemas de control tipo PLC, sensores lógicos y realizan la medición de distintos parámetros en segundos o fracciones del mismo (espesor, longitud, diámetro, concentricidad y temperatura). Este equipamiento funciona como control de calidad y garantiza la aceptación del producto terminado, y es operado en su totalidad a través

de una computadora que fallas de manera temprana. Funciona de manera ininterrumpida las 24 horas del día.

5.2-Los procesos de cambio.

Resulta de interés entonces, pensar en la industria de hoy y en los riesgos que representan los procesos de cambio, debido al empleo de tecnología más moderna, modificación en los procedimientos, mejoras de calidad o adecuación a nuevas regulaciones y su impacto en los operadores. Este proceso tiene aspectos negativos y positivos. (Fig. 5)

Esto quiere decir que la instalación necesita implementar un estudio que contemple los cambios necesarios para evitar la ocurrencia de accidentes o fallas, y en este caso se debe seguir un plan que contemple:

- ✓ Detectar los sitios donde puedan ocurrir fallas, realizando un diagnóstico temprano.
- ✓ Analizar que originó la falla.
- ✓ Buscar información (antecedentes).
- ✓ Evaluar quienes intervinieron en la falla.
- ✓ Elegir el método más eficiente para resolverla.
- ✓ Elaborar un plan de acción con las respectivas prioridades.
- ✓ Realizar las tareas de seguimiento.

Aspectos negativos	Aspectos positivos
Puede haber confusiones	**Adaptación a la nueva instancia**
Resistencia a lo nuevo	**Desarrollo de creatividad**
Puede haber desconcierto	**Mayor capacitación**
Lo consideran una imposición	**Mejor "performance" en la tarea**

Figura 5. El efecto del cambio y la modernización.

6-Conclusiones.

Todos los comentarios realizados hasta aquí, son aplicables a cualquier industria o proceso industrial que utilice material radiactivo en la actualidad, independientemente del uso que tenga el medidor industrial o el sistema de medición empleado.

De las expresiones vertidas en este trabajo se desprende que hay responsabilidades compartidas entre la instalación y el responsable.

La instalación debe:

- **Evaluar las condiciones de aptitud del responsable por la seguridad radiológica, su formación y percepción del riesgo radiológico.**
- **Analizar el crecimiento técnico y su preparación. (Entrenamiento previo, experiencia, habilidad técnica, nivel de comunicación, liderazgo, etc.).**
- **Capacitarlo en forma continua y actualizar sus conocimientos.**
- **Evitar que el mismo tenga una carga de trabajo que le impida cumplir correctamente con sus funciones.**
- **"Motivar" o "incentivar" al responsable a implementar mejoras en cuanto a calidad se refiere y apoyar el crecimiento personal y grupal.**

El responsable debe:

- **Evaluar y cumplimentar acciones tendientes al control radiológico permanente del material radiactivo a su cargo.**

- **Detectar posibles fallas de los sistemas empleados de manera preventiva.**
- **Elaborar procedimientos con pautas claras y efectivas.**
- **Desarrollar su tarea con "actitud" y "conducta" durante todo el ejercicio de su función.**
- **Buscar permanentemente la eficiencia y la eficacia.**

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10 Years of Cooperation between the Department of Energy and the Nuclear Regulatory Authority of Argentina

Manning, M.; Hayes, S.; Valentino, L.I.; Gariazzo, C.; Bonino, A.D.; Whitaker, M. and Glidewell, D.

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10 YEARS OF COOPERATION BETWEEN THE DEPARTMENT OF ENERGY AND THE NUCLEAR REGULATORY AUTHORITY OF ARGENTINA

[1]Manning, M.; [2]Hayes, S.; [3]Valentino, L.I.; [2]Gariazzo, C.; [3]Bonino, A.D.;
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ASTRACT

The Department of Energy (DOE) and the Nuclear Regulatory Authority (ARN) of Argentina have collaborated together over the past 10 years through the (U.S. DOE-ARN) Safeguards Cooperation Agreement to strengthen the international regime. The agreement was established initially to cooperate in nuclear material control, accountancy, physical verification of the nuclear material, physical protection and new trends in containment and surveillance technologies for international safeguards applications.

The collaboration has focused on exploring methods of strengthening the international safeguards regime by developing a methodology and software for applying safeguards to a gaseous diffusion uranium enrichment plant, evaluation and exploration of surveillance and containment technology, development of sensors for the detection of nuclear materials in a spent fuel site, remote monitoring used for controlling the spent fuel transfers, environmental sampling, and the development of a neutron calibration center. Many of the initial collaborative efforts were evaluations of new technologies and how they could be utilized to provide confirmation of nuclear materials thereby strengthening the international safeguards. This paper will discuss the process of evaluation of these technologies and the experiences gained during this 10-year collaborative effort.

INTRODUCTION:

ARGENTINA AND USA - LEGAL FRAMEWORK:

Argentina signed the Treaty for the Prohibition of Nuclear Weapons in Latin America and the Caribbean (Treaty of Tlatelolco) on 27th September 1967, and the Treaty for the Non Proliferation of Nuclear Weapons (NPT) in 1968. Argentina also signed the Agreement Between the Republic of Argentina and the Federative Republic of Brazil for the exclusively Peaceful Use of Nuclear Energy in 1991. In 1994 it was signed the Agreement between the Republic of Argentina, the Federative Republic of Brazil, the Brazilian Argentine Agency for Accounting and Control of Nuclear Materials and the International Atomic Energy Agency for the Application of Safeguards.

The Agreement between the National Atomic Energy Commission of Argentina and the U.S. Department of Energy Concerning Research and Development in Nuclear Material Control, Accountancy, Verification, Physical Protection, and Advanced Containment and Surveillance Technologies for International Safeguards Applications was signed in Bariloche in 1994, Argentina.

In 1994 it was also signed the Agreement to Extend and Amend the Agreement between the Nuclear Regulatory Authority of Argentina and the United States Department of Energy concerning Research and Development in Nuclear Material Control, Accountancy, Verification, Physical Protection, and Advanced Containment and Surveillance Technologies for International Safeguards Application.

In addition, the United States (U.S.) was in the process of providing access to some of its facilities for safeguards as part of a Voluntary Offer Agreement to the International Atomic Energy Agency (IAEA), in which the U.S. would have designated facilities on a list, in which the IAEA could choose to implement safeguards on facilities which would that contained materials excess to the U.S. national security needs. As part of the effort, the U.S. wanted to demonstrate

through transparency of safeguards being implemented on U.S. facilities that they did not have an economic advantage over other non-weapon member states. The U.S. also wanted to collaborate with Argentina in helping to design and develop methods for verification as the 93+2 policy measures were being evaluated by the IAEA and Member States to help to strengthen the Nonproliferation Treaty requirements by allowing the Agency to have access to facilities to implement environmental sampling.

HISTORICAL COLLABORATION PRIOR TO THE FORMAL AGREEMENT

The initial collaboration was provided in the areas of nondestructive assay measurements (NDA) for hold-up in process streams, such as the enrichment facilities, and other nuclear material control areas.

The United States Department of Energy (DOE) and the Nuclear Regulatory Authority of Argentina began cooperating in December 1992 to evaluate different methods for measuring materials held up in process streams.

The U.S. in collaboration with Argentina conducted a workshop on the implementation of safeguards at a gaseous diffusion facility. The Argentines conducted tours of feed plants at the Portsmouth Gaseous Diffusion Plant and toured the transfer product facility. NDA measurement techniques were discussed to address process inventory and how measurements were implemented at the X-333 site. Discussions were made on how non-destructive uranium enrichment determination in process hold-up deposits were evaluated and accounted for in the process inventory. This was the beginning foundation on which the U.S. worked together with the Nuclear Regulatory Authority to sign an agreement for safeguards purposes.

SIGNATURE OF THE DOE-ARN SAFEGUARDS AGREEMENT

The Agreement to Extend and Amend the Agreement between The Nuclear Regulatory Authority of Argentina and the United States Department of Energy Concerning Research and Development in Nuclear Material Control, Accountancy, Verification, Physical Protection and Advance Containment and Surveillance Technologies for International Safeguards Applications was signed 18th April 1994 in Argentina.

PURPOSE OF THE AGREEMENT

Cooperation under this Agreement may include but it is not limited to:

- Exchange of information, equipment, funding or personnel
- Exchange or loan of materials, equipment, and components for evaluation and testing
- Joint projects for the research, development, testing and evaluation of nuclear material control, accountancy, verification, physical protection and advance containment and surveillance technologies, techniques or procedures.

COLLABORATIVE ACTIVITIES

The initial collaborative activities were mostly based on Traditional Safeguards. An instance of these activities were the ones related to the Material Control Accountancy (MCA) and Surveillance Technology. The activities were formalized in the format of a document named Action Sheet (AS), which describe the objectives, the cooperative activities to be carried out by each of the signing parts and also the Projects Leaders.

Later collaborative activities include:

1. Remote monitoring

As part of the International Remote Monitoring Project (IRMP) field trials, during the month of March, 1995 a Remote Monitoring System (RMS) was installed at the Embalse Nuclear Power Plant in Córdoba province, Argentina. This system monitors the status of four

typical Candu spent fuel storage silos. The monitoring equipment for each silo consisted of analog temperature and gamma radiation sensors and digital motion and electronic fiber-optic seals connected to a wireless Authenticate Item Monitoring System (AIMS).

In 1998 a RMS consisting of six gamma and one neutron radiation sensors was installed at the Embalse Nuclear Power Plant. Five gamma sensors utilized RF transmission to communicate with Echelon nodes connected to a Local Operating Network (LON). One gamma and one neutron sensor were hardwired to the LON network. The system was developed for Los Alamos, Sandia, Oak Ridge and ARN laboratories. The primary goal of this system was to demonstrate that continuity of knowledge could be maintained throughout the transfer process from the spent fuel pond to the silos field using radiation sensors to monitor the transfer path. The system development went through a series of modifications to meet IAEA requirements prior to final installation at Embalse NPP. System requirements and a formal test plan were generated and tests were conducted at SNL, Los Alamos, and at the IAEA.

2. **Physical Protection:** In this framework a workshop on the Design Basis Threat has been given in April, 2005 and also a Regional Course on Physical Protection of Nuclear Materials and Facilities will be given in next September. Besides, it is being exchanged technical experience and knowledge on the implementation and application of the methodology on Vital Area Identification and Sensor Testing in extreme environments (eg. Fog, humidity, etc.)

3. **Neutron Cablibration Center:** A Neutron Callibration Center is being installed to be used for training purposes for safeguards inspectors from ARN, ABACC and IAEA in NDA measurement by passive and active interrogation methods.

4. Virtual Private Network (VPN): This activity is performed to develop and implement a Nuclear Regulatory Authority Safeguards Network . A workshop on Private Networks was conducted in December 2002 with experts from the Sandia National Laboratories (SNL). A tunnel was established as a test between ARN laboratories at Ezeiza Atomic Center, B.A and ARN Southern Regional Office at Bariloche.

INTEGRATION OF SAFEGUARD ACTIVITIES

5. Environmental Sampling

This Action Sheet (AS) was signed in 1996, under the title of “Environmental Sampling”, which comprises the taking of Environmental Samples and “Swipe Samples”. For safeguard purposes, it was only considered the samples “Swipe”.

The wrapping up of this AS is associated to the certification of the laboratories involved which perform determinations through specific techniques.

The main objectives of the “Environmental Sampling” AS are:

- To have a procedure to be used for national verification purposes
- To form part of the laboratory network of the IAEA
- To satisfy the technical analysis capability required for ABACC

This AS was divided in two phases. The first one consisted of an interchange of experiences to establish a Quality Assurance Program (QAP) to obtain the Laboratories Certification for analysis of swipe samples. The global process comprises from the decision making of taking this kind of sample in some facilities up to the evaluation of the results of the analysis obtained. This implies to define safeguards objectives to decide to take swipe samples, to recover the specific facility information, to program the swipe sample campaign, the swipe sampling taking, to define the laboratory techniques to make the analysis and the results evaluation.

The flow sheet below describes the activities to be performed and shows the different steps of the general process the samples will undergo to obtain the final results.

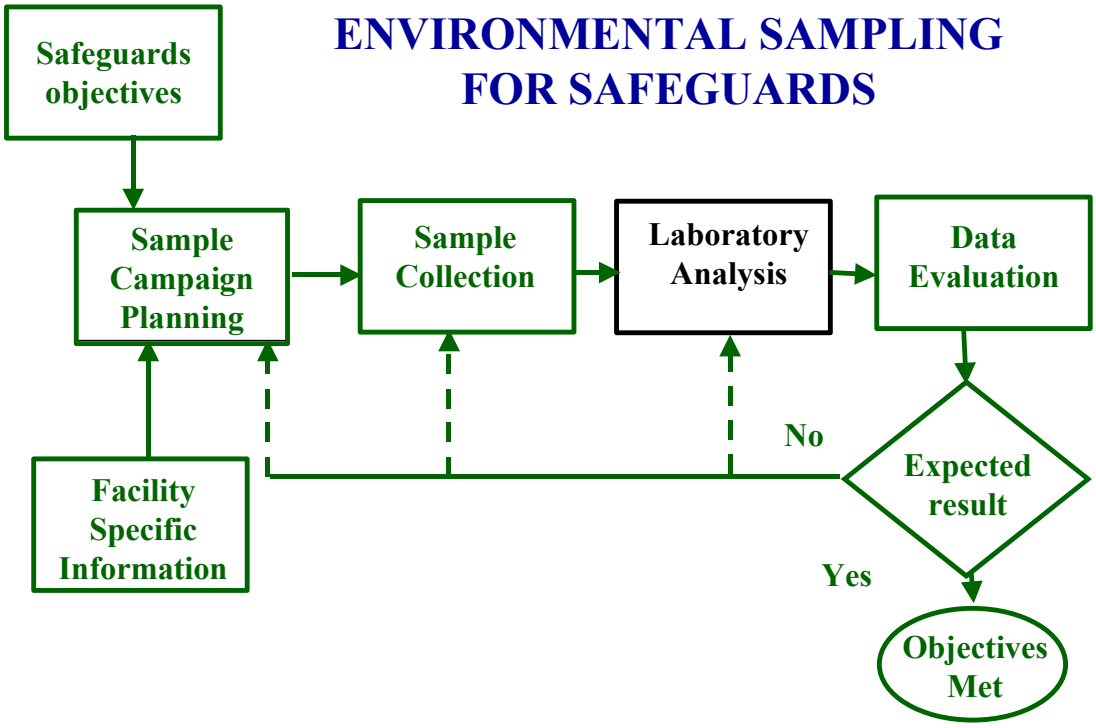
The second phase to fulfil these AS is to perform an inter-comparison exercise for analysis of swipe samples.

The resulting Quality Assurance Handbook (QAH) will be translated into English and submitted, to receive advice and interchange of experience with the DOE, to certify the process. The use of a specific software to evaluate the results is being evaluated.

The QAH will consist of:

- Procedure of the swipe samples campaign.
- Instructions to take swipe samples
- Procedure to use measurements techniques to detect U and Pu for mass spectrometry.
- Documentation associated to the Planning Campaign, swipe sampling taking and the sample shipments to the laboratories.

FUTURE ACTIVITIES



- **Software for Control of Nuclear Material**

To develop a software to register and inform in a controlled and safety manner the shipment - receipt and any other inventory change (IC) of nuclear material (NM) among facility operators distributed along the country. This nationally could change the accounting verification dynamics and philosophy as it would be possible to make the inventory audit of the NM from the Headquarters instead of at the facilities. The soft should be developed in compliance with safeguards national and international standards and requirements.

This software should be a tool for all the operators at the facilities and the National Authority. This system will permit to obtain information of the inventory in each facility in real time and the flow between facilities, to make crosscheck, for consistency.

The system should be secure enough to assure that the inventories of NM at each facility are confidential and could not be either seen by operators of other facilities or modified by any other that the licensed operator responsible for a Material Balance Area and the NM.

- **International Safeguards and Security Training and Support Centre in Argentina**

To develop a Safeguards and Security Training and Support Centre (SSTSC) at a site to be determined in Argentina. The Centre will offer training and technical support for safeguards and security integrated hardware systems; training for practices and procedures related to safeguards and security of nuclear and radiological materials and training and managements support for programmatic safeguards and security concerns. It will also allow for the integration and demonstration of prototype physical protection system configurations before actual field deployment.

- **Cooperation on Nuclear Export Controls**

To curb the proliferation of weapons of mass destruction (WMD) and their delivery systems by strengthening associated export control systems and practices.

The objective is to grow and improve national systems of export control, through joint programs in enforcement, licensing and industry outreach.

The general task to be complied is:

- ✓ Nuclear Commodity Identification Training (CIT) Program to familiarise customs inspectors and others associated with the management of export – controlled items.

- ✓ Nuclear Export Control Technical Exchange: sharing experience and best practices related to nuclear export controls through a technical exchange meeting of Argentina and the US government and technical personnel involved in nuclear export control.

- ✓ Nuclear Industry Outreach Cooperation: enhance export control awareness and compliance of Argentine nuclear and nuclear related enterprises through conducting a nuclear export control and internal compliance workshop.

CONCLUSIONS:

Later activities were focused on how safeguards could be strengthened through the implementation of technologies being considered as part of the Additional Protocol measures.

The cooperation would help to evaluate technologies that would employ remote monitoring for continuity of knowledge of materials through out the spent fuel process, environmental sampling for additional verification of materials not be diverted or reprocessed, and strengthened physical protection measures.

Over the past 10 years the collaboration has been a useful vehicle through which new technologies were evaluated and field tested in facilities. Some of the initial demonstrations were not completely deployed over to the Agency until new avenues were developed in which large sums of data could be transferred through a secure means. These did not indicate failure, but instead provided an opportunity to evaluate how continuity of knowledge for spent fuel could be monitored. In addition, there were some great successes such as the methodology for implementation of safeguards at a gaseous diffusion facility could be accomplished.

The cooperation has been effective at addressing emerging issues and serves as a means by which countries can work together to achieve common goals of strengthening the nonproliferation by evaluating technologies to reach policy objectives in a cooperative and transparent manner.

Prevención como objetivo principal en las prácticas regulatorias relativas a reactores de investigación

Waldman, R.M.

Presentado en: Taller Regional sobre Reactores de Investigación.
Lima, Perú, 1 al 5 de agosto de 2005

PREVENCIÓN COMO OBJETIVO PRINCIPAL EN LAS PRÁCTICAS REGULATORIAS RELATIVAS A REACTORES DE INVESTIGACIÓN

Waldman, R.M.

Autoridad Regulatoria Nuclear
Argentina

INTRODUCCIÓN

En la Argentina, los propósitos de uso de los reactores de investigación y conjuntos críticos son muy diversos, variando desde la producción de radioisótopos a la investigación o la docencia. También son diversas las características de los Titulares de las Licencias, yendo desde La Comisión Nacional de Energía Atómica, hasta Universidades Nacionales.

La siguiente Tabla resume algunos datos de los reactores de investigación y conjuntos críticos en la Argentina:

Reactor	Titular de Licencia	Tipo	Potencia	Tipo EECC	Enriq.	Sistemas de extinción	Exceso de reactividad
RA-0	Univ. Córdoba	CC	1 W	Barras cilíndricas	< 20%	Barras absorbentes Desag. moderador	0,4 $
RA-1	CNEA	RI	40 kW	Barras cilíndricas	< 20%	Barras absorbentes	1,2 $
RA-3	CNEA	RI	10 MW	placas MTR	< 20%	Barras absorbentes	~ 8 $
RA-4	Univ. Rosario	CC	1 W	Discos homogéneos	< 20%	Barras absorbentes Separación	0,4 $
RA-6	CNEA	RI	500 kW	placas MTR	90%	Barras absorbentes	~ 2,5 $
RA-8	CNEA	CC	10 W	Barras cilíndricas	< 3.4%	Barras absorbentes Desag. moderador	

Tabla I: **Datos de Reactores de Investigación y Conjuntos Críticos en Argentina**

Ante este variado panorama, y teniendo en cuenta el riesgo potencial asociado a la operación de cada instalación, la estrategia utilizada para el control regulatorio se basa especialmente en la PREVENCIÓN.

PREVENCIÓN, LA ESTRATEGIA REGULATORIA MÁS IMPORTANTE

La PREVENCION se logra desde la perspectiva regulatoria, mediante la existencia de:

- *un Marco regulatorio* adecuado,
- *herramientas regulatorias* apropiadas,
- una *actitud pro-activa hacia la seguridad*,
- *personal competente* para hacer su tarea,
- una *organización adecuada*.

Marco regulatorio

Existen leyes nacionales que definen las responsabilidades y funciones de la Autoridad Regulatoria Nuclear (ARN), y de los Titulares de Licencia (o Entidades Responsables) de los reactores nucleares.

Las responsabilidades y funciones de ambas instituciones son compatibles con las recomendadas por el Código de Conducta para Reactores de Investigación.

Respecto del marco regulatorio y en lo relacionado con los reactores nucleares, es responsabilidad de la ARN:

- Emitir normas.
- Licenciar instalaciones.

- Licenciar el personal.
- Aplicar el régimen de sanciones y multas.

Emitir normas y guías

Las normas argentinas tienen en cuenta aspectos determinísticos y probabilísticos. No se utiliza el concepto de accidentes bases de diseño y fuera de la base de diseño, sino que considera que todos los accidentes son posibles y que el diseño de una instalación debe cumplir con un criterio cuantitativo (curvas criterio para los trabajadores y el público). La ARN verificó el cumplimiento de este criterio para dar su opinión sobre el diseño del reactor de Australia.

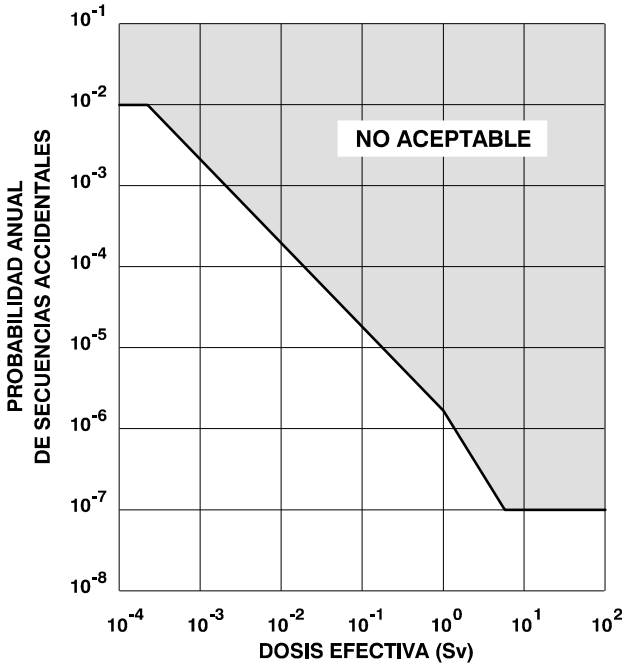


Figura N°1 - **Curva Criterio para el Público**

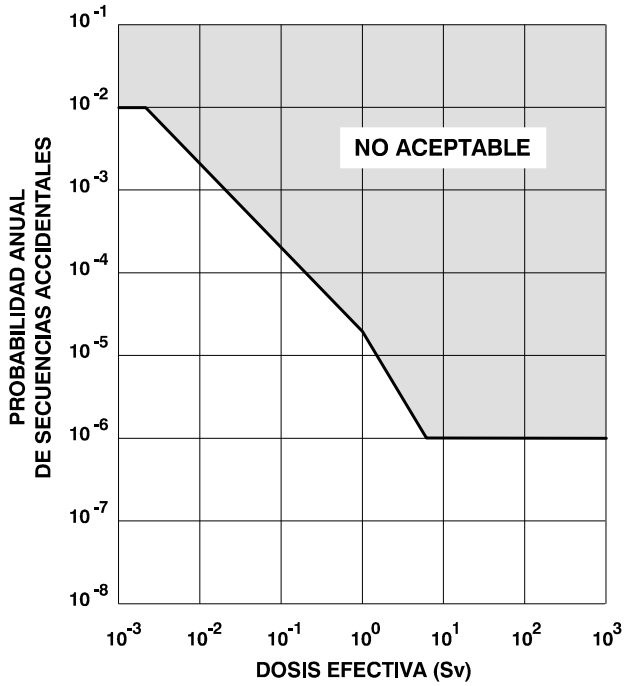


Figura N°2 - **Curva Criterio para los Trabajadores**

En el proceso de elaboración de las normas y guías se tienen en cuenta las recomendaciones internacionales y la experiencia operativa nacional e internacional.

Existe un mecanismo de consulta a los destinatarios de aplicación y al público, en forma previa a la aprobación formal de nuevas normas o modificaciones a las ya existentes. La consulta no es vinculante, y su propósito es recabar información sobre su factibilidad técnica y las consecuencias que se derivarían de su implementación.

Hay normas y guías específicas para reactores de investigación y también para conjuntos críticos (uno de los pocos países del mundo). En el Anexo I se adjunta el listado de las normas y guías aplicables a dichas instalaciones.

La normativa argentina establece que el Jefe del Reactor es el Responsable Primario de la Instalación y requiere la constitución de dos Comités para tratar aspectos relacionados con la seguridad radiológica y nuclear:

- Comité de Revisión Técnica para asesorar al Titular de Licencia, independiente de la instalación, y que debe estar constituido por profesionales con amplia experiencia en la operación de reactores nucleares y ser aceptados por la ARN.

  El Comité debe reunirse periódicamente para revisar el desarrollo de la operación del reactor desde el punto de vista de la seguridad y de la protección de las personas, incluyendo el análisis de incidentes operacionales y otros sucesos relevantes que hubieran ocurrido y la evaluación de las propuestas que el Responsable Primario o Titular de Licencia hiciese acerca de modificaciones al diseño de la instalación o de nuevos experimentos o irradiaciones que involucren al reactor.

- Comité Interno de Seguridad para asesorar al Responsable Primario, formado por personal de la instalación y otros externos.

  Los miembros de este Comité deben ser designados por el Responsable Primario y la composición del mismo debe ser tal que los miembros que pertenecen al plantel de la instalación no deben superar el 50%, en tanto que los restantes deben pertenecer a otros sectores de la organización o ser expertos independientes. Este Comité trata aspectos de seguridad radiológica y nuclear que surgen durante la operación del reactor.

La normativa argentina fue utilizada para el licenciamiento de los reactores RA 0, RA 6 y RA 8, para autorizar las modificaciones significativas realizadas al reactor RA 3 en la marco del Programa de Modernización y Aumento de Potencia y para autorizar nuevos experimentos (p.e. irradiar EECC prototipos de U-Si). Asimismo las normas fueron ampliamente vinculantes en el diseño de los reactores construidos por empresas argentinas en otros países: Perú, Argelia, Egipto y Australia, en los cuales la ARN opinó, al menos, sobre la licenciabilidad del diseño. En algunos de estos casos también se utilizó el criterio de licenciamiento de personal establecido por la normativa argentina y la ARN ayudó en parte del proceso.

Cabe mencionar que la normativa argentina relacionada con la operación de reactores de investigación tiene actualmente vigencia para el reactor ET-RR2 de Egipto.

Otro aspecto a destacar en la normativa argentina es que, con el objeto de mejorar la seguridad, se establecieron criterios cuantitativos para la reactividad asociada a los sistemas de extinción y de control de la reactividad y a los experimentos.

Las razones de seguridad para limitar al exceso de reactividad son las siguientes:

- *En los conjuntos críticos, los accidentes de reactividad son los que pueden producir las mayores consecuencias radiológicas . La normativa argentina contempla aplicar una limitación de 0.4$ al exceso de reactividad. Con esta limitación se evita la posibilidad de los llamados pulsos de potencia y se garantiza que el período de crecimiento de la potencia sea superior a 10 s (con lo cual la actuación de cualquier sistema de extinción, rápido o lento, garantiza la extinción del reactor).* La normativa

permite construir conjuntos críticos con excesos de reactividad mayores, pero establece un límite máximo a las dosis que pueden recibir los trabajadores en cualquier puesto de trabajo, en caso de ocurrencia de una excursión crítica que produzca 10^{18} fisiones.

- *Para los reactores de investigación de baja potencia (de algunos kW) la normativa contempla limitar el exceso de reactividad a un valor que no supere 1.5 $. Con esta limitación se garantiza la integridad de los EE.CC, aún ante la ocurrencia del accidente con el mayor exceso de reactividad (y de esta forma se evita la falla de los EECC y su efecto en las dosis que recibiría el personal).*

Licenciar instalaciones

El proceso regulatorio contempla la emisión de Licencias de Construcción, Puesta en Marcha, Operación y Retiro de Servicio. No existe una Licencia especial para el sitio, ya que para el otorgamiento de la Licencia de Construcción, se evalúa la interacción instalación-sitio.

Las licencias de Operación para los reactores nucleares tienen una validez máxima de cinco años y para su renovación el Titular de Licencia debe presentar a la ARN, para su revisión y aprobación, una evaluación integral de la seguridad de la instalación.

Como parte del proceso de licenciamiento de nuevas instalaciones, está bajo análisis en la ARN, la aplicación de un mecanismo de consulta pública no vinculante.

Licenciar el personal

La ARN otorga dos tipos de documentos para el licenciamiento del personal que opera un reactor nuclear y que implican certificaciones conceptualmente diferentes:

- *Licencia Individual*: Es un certificado de carácter permanente que reconoce la capacidad técnico-científica necesaria para ejercer una determinada función dentro del Organigrama de Operación de un determinado tipo de Instalación (p.e. Jefe de Reactor para Conjuntos Críticos). Para su otorgamiento se requiere demostrar capacitación básica, capacitación especializada y experiencia laboral.
- *Autorización Específica*: es un certificado renovable que tiene una validez máxima de dos años y que habilita a una persona licenciada a ejercer dicha función en una instalación particular. Para su otorgamiento se requiere poseer una Licencia Individual apropiada para la función y se evalúa la capacitación específica, el entrenamiento en la función, el re-entrenamiento y la aptitud psicofísica.

Aplicar Sanciones y Multas

Para el caso de un incumplimiento de lo establecido en la documentación de carácter mandatorio (Licencia, normas, Informe de Seguridad, Manual de Operaciones, etc.) la ARN puede aplicar el régimen de sanciones, que contempla según la gravedad del caso: apercibimiento, multas, suspensión de una Licencia o Autorización Específica o su revocación.

El régimen de sanciones funciona como el último eslabón de la cadena de seguridad. En efecto, si el sistema es realmente efectivo y si los Titulares de Licencia ejercen plenamente sus responsabilidades, la aplicación de sanciones debería solo ocurrir en casos excepcionales. Lo contrario indicaría, entre otras cosas, un pobre comportamiento regulatorio. En tal sentido una función regulatoria no formal de la ARN es concientizar a los Titulares de Licencias y a los Responsables Primarios sobre sus responsabilidades por la seguridad, y que cada vez hagan más suya la aplicación de la Cultura de la Seguridad.

Herramientas utilizadas para el control regulatorio

El control regulatorio se realiza principalmente mediante inspecciones, auditorías y evaluaciones de seguridad. Además se realiza un estricto control de las dosis

ocupacionales, de las descargas de material radiactivo al ambiente, de las dosis en el público y la vigilancia ambiental en los alrededores de las instalaciones nucleares.

La actividad regulatoria en relación a las exposiciones potenciales está basado en la prevención y en la preparación para la mitigación. La prevención se aplica desde la etapa de diseño y construcción de la instalación y continua durante la operación. La actividad relacionada con la mitigación se realiza contemplando la exigencia de los sistemas de seguridad como así también actuar sobre la situación posterior al accidente a través de los Planes de Emergencia.

Las inspecciones y auditorías son de tres tipos:

- inspecciones rutinarias que contemplan temas tales como protección radiológica, mantenimiento y operación.
- Inspecciones no rutinarias. Estas se desarrollan ante la ocurrencia de eventos, para presenciar las pruebas de puesta en marcha asociadas a modificaciones a la instalación o nuevos experimentos, etc.
- Inspecciones especiales que se realizan al inicio del período de operación anual con el objetivo de realizar el mantenimiento necesario con la instalación parada, probar el funcionamiento de los sistemas de seguridad y verificar las condiciones de “núcleo operable”.

Las inspecciones y auditorias tienen como objetivo verificar que en la instalación:

- Se cumplen las condiciones establecidas en la documentación de carácter mandatorio.
- Se mantienen las funciones de seguridad establecidas por diseño para las estructuras, componentes y sistemas de la instalación.
- Se dispone en todo momento con el personal necesario y con la calificación requerida para la ejecución de las tareas de control regulatorio.
- Se aplican adecuadamente los principios de cultura de seguridad.
- Se realizan las acciones correctivas para subsanar y evitar su reiteración, en las deficiencias y las condiciones anormales detectadas.

Al comienzo del año se planifica un programa de inspecciones rutinarias y auditorías. El alcance de este programa y las frecuencias de las inspecciones contempla tanto el riesgo potencial postulado para la instalación como el cumplimiento del objetivo mencionado previamente.

Para la implementación del programa se hace uso de distintas metodologías que pueden agruparse en:

- Observación directa de la ejecución de tareas.
- Entrevistas con el personal.
- Verificación y control de procedimientos, registros y documentación.
- Verificar el entrenamiento específico del personal.
- Verificar “in situ”, cuando sea posible, los valores de disparo y la actuación de los sistemas de seguridad.
- Realizar, cuando sea posible, tomas de muestras, pruebas, verificaciones y mediciones independientes (especialmente en el área radiológica).

Hay un “check-list” para cada instalación que se utiliza como ayuda para la preparación y la ejecución de las inspecciones.

La tabla siguiente resume el esfuerzo regulatorio contemplado en el programa correspondiente al año 2005. En la columna correspondiente a los días personas se muestra el esfuerzo regulatorio “in-situ” y no incluye los tiempos de preparación ni de redacción de los informes de inspección.

INSTALACION	N° Insp. y Audit.	Días-persona
RA-0	4	24
RA-1	6	48
RA-3	12	102
RA-4	4	24
RA-6	6	54
RA-8	2	2
TOTALES		**254**

Se realizan dos tipos de evaluaciones de seguridad:

- Evaluaciones rutinarias (descarga de efluentes, manejo de residuos, dosimetría del personal, informes de los comités de seguridad, etc.).
- Evaluaciones no rutinarias (eventos relevantes, modificaciones significativas a la instalación o nuevos experimentos, actualización del informe de seguridad, gestión de núcleo, ejercicios de aplicación del plan de emergencias, cumplimiento de requerimientos regulatorios, etc).

El personal de la ARN dedicado al control de los reactores nucleares dispone del apoyo de especialistas en temáticas tales como: protección radiológica, medición de actividad, neutrónica, termohidráulica, mecánica, eléctrica, materiales APS, I&C y seguridad operacional (evaluación de eventos, cultura de seguridad, aseguramiento de la calidad) que también son utilizados para el control de las centrales nucleares.

Si durante las tareas de control regulatorio se detectan tendencias desfavorables o desvíos respecto a las condiciones normales de operación, se pueden aplicar acciones preventivas o correctivas tales como la emisión de requerimientos regulatorios.

En lo que respecta a la vigilancia ambiental, la intervención de la ARN consiste en el monitoraje ambiental alrededor de las instalaciones nucleares tomando muestras representativas de los diferentes compartimentos de la matriz ambiental de transferencia de radionucleídos y evaluando el impacto ambiental de las descargas líquidas y gaseosas.

Con tal propósito se toma muestras de agua, sedimentos, plancton, peces, agua potable, pastos, alimentos producidos en la zona, etc.

El impacto radiológico de los resultados obtenidos a partir del monitoraje ambiental, se determina mediante la estimación de las dosis en los miembros del público. Las dosis se calculan para el individuo promedio del grupo crítico, suponiendo que todos los alimentos que se consumen en la zona son de origen local.

La ARN cuenta con un Sistema de Intervención en Emergencias Radiológicas (SIER) para tratar emergencias radiológicas en instalaciones y prácticas menores o que involucre a la población. El mismo está vinculado al Sistema Federal de Emergencias y presta además asesoramiento a autoridades públicas y usuarios.

Actitud pro-activa

La existencia de una actitud pro-activa hacia la seguridad, es una de las fortalezas del accionar regulatorio. Dicha actitud se manifiesta por ejemplo en:

- Revisión continua de las recomendaciones internacionales y de la normativa vigente.
- Presencia frecuente de inspectores en las instalaciones.
- Participación activa en los mesas examinadoras de licenciamiento y re-entrenamiento del personal.
- Aplicación de aspectos de calidad y de mejora continua en la tarea regulatoria.

- Realización de reuniones periódicas con representantes de los Titulares de Licencia, con discusiones francas de los problemas existentes, como mecanismo adecuado para lograr mejoras en las instalaciones y en las organizaciones.
- Apoyo a la aplicación del Código de Conducta para Reactores de Investigación.

Personal competente

Es condición necesaria para cumplir adecuadamente con el control regulatorio el mantener la capacidad técnica del personal. Esto se puede lograr a través de un programa de capacitación continua del personal, que incluye las actividades docentes. Las actividades principales en lo relacionado con la capacitación son:

- Asistencia a los cursos de postgrado en protección radiológica y seguridad nuclear del personal profesional y técnico ingresante a la ARN.
- Asistencia a cursos de capacitación específicos.
- Participación en congresos y talleres en el ámbito nacional e internacional.
- Colaboración en actividades docentes a nivel nacional e internacional.
- Colaboración con otros órganos reguladores en temas específicos.
- Participación en la generación de documentación en las áreas de seguridad radiológica y nuclear. En particular en los documentos técnicos del OIEA.
- Participación en el IRSRR.

Organización

La organización debe ser tal que permita la transparencia y traceabilidad de la tarea regulatoria, donde queden claramente definidas las responsabilidades individuales y colectivas para poder realizar el control y existan mecanismos para encontrar y corregir deficiencias.

En el plano general esto significa la necesidad de planificar, realizar y controlar las actividades vinculadas con el accionar regulatorio de los reactores nucleares, y la planificación de los recursos económicos y humanos necesarios.

En el plano particular, las tareas se deben realizar siguiendo procedimientos específicos. En la ARN existen procedimientos para realizar las inspecciones, auditorías y evaluaciones, la comunicación de eventos, el control regulatorio durante la aplicación del ejercicios del plan de emergencia y la emisión de requerimientos regulatorios. En el Anexo II se da el listado de procedimientos aplicables a los reactores nucleares.

Se instauró la designación de los Responsables de Proceso como mecanismo para encontrar y corregir deficiencias.

Otros aspectos

- Actuación de los Comités de Revisión Técnica.
- Planes de Emergencia en Centros Atómicos.
- Programa de desmantelamiento.
- Evaluación de pares.
- Programa de envejecimiento.
- Aspectos de organización del Titular de Licencia.
- Falta de normas específicas sobre desmantelamiento.
- Falta de definición para el uso de códigos y normas industriales.
- Falta de aplicación del APS como herramienta regulatoria.
- Conocimiento de la población del rol de la Autoridad Regulatoria.

- Política de comunicación de incidentes a la opinión pública.
- Mantenimiento de la capacidad técnica.

Consejos

- Expresarse correctamente en documentos que incluyan temas de seguridad o de cumplimiento de la documentación de carácter mandatorio.
- Proponer límites y condiciones de operación adecuados y cumplibles.
- No interpretar en forma creativa cuando se debe suspender la operación de la instalación para dar cumplimiento a lo que establece la Licencia o la normativa.
- Preservar la independencia de la Autoridad Regulatoria y no comprometerla en problemas internos de la organización del Titular de Licencia.
- Cumplir con los aspectos formales en la interacción con la Autoridad Regulatoria.

Propuestas:

- Foro de reguladores aplicable a los reactores nucleares del área caribe y latino-américana.
- Programa de Desmantelamiento y normativa aplicable a los reactores nucleares del área caribe y latino-américana.
- Colaboración para la formación de personal en la realización del control regulatorio o en la utilización de los reactores nucleares.

ANEXO I

Listado de normas y guías regulatorias argentinas aplicables a los reactores de investigación y conjuntos críticos.

Normas Regulatorias

Código	Título de la norma	Revisión
AR 0.0.1.	Licenciamiento de instalaciones Clase I	Rev. 2
AR 0.11.1.	Licenciamiento de personal de instalaciones Clase I	Rev. 3
AR 0.11.2.	Requerimientos de aptitud psicofísica para autorizaciones específicas	Rev. 2
AR 0.11.3.	Reentrenamiento de personal de instalaciones Clase I	Rev. 1
AR 4.1.1.	Exposición ocupacional en reactores nucleares de investigación	Rev. 0
AR 4.1.2.	Limitación de efluentes radiactivos en reactores nucleares de investigación	Rev. 1
AR 4.1.3.	Criterios radiológicos relativos a accidentes en reactores de investigación	Rev. 2
AR 4.2.1.	Diseño de conjuntos críticos	Rev. 1
AR 4.2.2.	Diseño de reactores de investigación	Rev. 1
AR 4.2.3.	Seguridad contra incendios en reactores de investigación	Rev. 2
AR 4.5.1.	Diseño del sistema de suministro de energía eléctrica de reactores de investigación	Rev. 1
AR 4.7.1.	Cronograma de la documentación a presentar antes de la operación de un reactor de investigación	Rev. 1
AR 4.7.2.	Cronograma de la documentación a presentar antes de la operación de un conjunto crítico	Rev. 0
AR 4.8.1.	Pruebas preliminares y puesta en marcha de conjuntos críticos	Rev. 1
AR 4.8.2.	Pruebas preliminares y puesta en marcha de reactores de investigación	Rev. 1
AR 4.9.1.	Operación de conjuntos críticos	Rev. 1
AR 4.9.2.	Operación de reactores nucleares de investigación	Rev. 2
AR 10.1.1.	Norma Básica de Seguridad Radiológica	Rev. 3
AR 10.12.1.	Gestión de residuos radiactivos	Rev. 1
AR 10.13.1.	Transporte de materiales radiactivos	Rev. 1

Guías Regulatorias

GR-4	Diseño de reactores nucleares de investigación	Rev. 0
GR-7	Diseño de conjuntos críticos	Rev. 0

ANEXO II

Listado de procedimientos aplicables al control regulatorio de los reactores de investigación y conjuntos críticos

Código	Revisión	Título del Documento
G-0XX-02	Rev. 1	Elaboración y revisión de normas y guías regulatorias
PP-UPP-001	Rev. 0	Elaboración del Plan de Trabajo y Presupuesto
PP-GSRN-001	Rev. 0	Registro, evaluación y comunicación de eventos al INES
PP-GSRN-002	Rev. 0	Emisión y control del cumplimiento de requerimientos regulatorios en instalaciones clase I
IT-GSRN-001	Rev. 0	Evaluación de dosis
IT-SGRN-001	Rev. 0	Comunicación de eventos relevantes en reactores nucleares argentinos
PP-SGRN-004	Rev. 0	Gestión de eventos relevantes en reactores nucleares
PP-SGRN-006	Rev. 0	Inspecciones y auditorías técnicas a reactores de investigación y conjuntos críticos en operación
PP-SGRN-007	Rev. 0	Tareas regulatorias relacionadas con los ejercicios anuales de aplicación de planes de emergencia de reactores de investigación y conjuntos críticos
PP-SGRN-009	Rev. 0	Evaluaciones rutinarias en reactores de investigación y conjuntos críticos
PP-SGRN-011	Rev. 0	Evaluaciones no rutinarias en reactores nucleares
En elaboración	.	Licenciamiento de los Reactores Nucleares
En elaboración		Fiscalización del entrenamiento y re-entrenamiento del personal de reactores de investigación y conjuntos críticos
PR-002	En Rev.	Intervención en emergencias radiológicas, Sistema SIER

Criterios de aceptación de riesgo de la Autoridad Regulatoria Nuclear

Felizia, E.R.

Publicado como PI 1/05 de la Autoridad Regulatoria Nuclear

CRITERIOS DE ACEPTACIÓN DE RIESGO DE LA AUTORIDAD REGULATORIA NUCLEAR

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RESUMEN

En este informe se describen algunas de las funciones de regulación y control que la ley atribuye a la Autoridad Regulatoria argentina en materia de riesgo radiológico y se presenta un análisis crítico de los criterios de aceptación del riesgo radiológico del sistema regulatorio argentino. Se reseñan las aplicaciones de las normas regulatorias AR 3.1.3. - "Criterios radiológicos relativos a accidentes en reactores nucleares de potencia" y AR 4.1.3 - "Criterios radiológicos relativos a accidentes en reactores de investigación" a casos concretos y se discuten los aspectos favorables y desfavorables de los criterios de aceptación de riesgo. Se concluye que el sistema regulatorio argentino cuenta con criterios de aceptación del riesgo radiológico adecuados y consistentes con los principios de protección radiológica de las personas y que no es conveniente por el momento introducirles modificaciones que amplíen la base conceptual en la que estos criterios se fundamentan.

ABSTRACT

This report describes some of the regulatory and control functions legally conferred upon the Argentine Nuclear Regulatory Authority concerning radiological risks, as well as a critical analysis of the radiological risk acceptance criteria contained in the Argentine regulatory system. A summary is made of the application of regulatory standards AR 3.1.3. - "Radiological criteria related to accidents in nuclear power reactors" and AR 4.1.3. - "Radiological criteria related to accidents in research reactors" to concrete cases, while the favourable and unfavourable aspects of the risk acceptance criteria are discussed. The conclusion is that the Argentine regulatory system contains adequate radiological risk acceptance criteria, that the latter are consistent with the radiological protection principles applicable to man and that, for the moment, there is no need to perform any modifications that would broaden the conceptual framework on which such criteria are based.

1. INTRODUCCIÓN

El Artículo 8b de la Ley Nacional de la Actividad Nuclear (Ley N° 24804) establece que la Autoridad Regulatoria Nuclear debe velar por la seguridad radiológica y nuclear en las actividades nucleares desarrolladas en la República Argentina. El Artículo 9a de la misma ley establece que toda persona física o jurídica, (Titular de Licencia), que desarrolle una actividad nuclear debe ajustarse a las regulaciones impartidas por la Autoridad Regulatoria y solicitar el otorgamiento de la licencia que lo habilite para su ejercicio. El otorgamiento de la licencia requiere - entre otros requisitos – que el Titular de Licencia demuestre el cumplimiento, por parte del diseño de la instalación para la cual éste solicita una licencia, de los criterios de aceptación del riesgo radiológico en exposiciones potenciales, establecidos en las normas correspondientes. En este trabajo se describen algunas de las funciones de regulación y control que la ley atribuye a la Autoridad Regulatoria en materia de riesgo radiológico y se presenta, además, un análisis crítico de los criterios de aceptación del riesgo radiológico del sistema regulatorio argentino

2. DESARROLLO

El riesgo radiológico impuesto a las personas por fuentes de radiación asociadas a una práctica ha sido considerado en las recomendaciones de instituciones internacionales como, por ejemplo, la Comisión Internacional de Protección Radiológica (CIPR) y la Organización Internacional de Energía Atómica (OIEA). Por ejemplo, en la publicación ICRP-60 de la primera de las instituciones mencionadas se recomienda que, en todo sistema de protección radiológica, "*el número de personas expuestas y la probabilidad de que se produzca una exposición, cuando no se tenga certeza de que ésta vaya a ocurrir, deberían mantenerse tan bajos como sea razonablemente alcanzable, teniendo en cuenta factores económicos y sociales*".

En el país existe, prácticamente desde el inicio de las actividades de la Comisión Nacional de Energía Atómica (CNEA), un sistema regulatorio que ha incorporado los criterios de aceptación del riesgo radiológico en caso de accidentes. En efecto, las normas AR 3.1.3. - "Criterios radiológicos relativos a accidentes en reactores nucleares de potencia" y AR 4.1.3 - "Criterios radiológicos relativos a accidentes en reactores de investigación", establecen las condiciones generales que debe cumplir el diseño de reactores nucleares de potencia y de investigación respectivamente, para prevenir la ocurrencia de accidentes y mitigar sus consecuencias radiológicas en el caso que estos ocurran. La norma AR 10.1.1. – "Norma básica de seguridad radiológica", sección D4. Exposiciones potenciales, establece criterios similares para la aceptación del riesgo radiológico de una instalación Clase I. El riesgo radiológico al que aluden estas regulaciones responde a un concepto de riesgo adoptado por la Autoridad Regulatoria argentina (Anexo I) y vigente desde hace tres décadas.

La norma AR 3.1.3 fue utilizada para evaluar el diseño conceptual de la central nuclear Atucha II, única aplicación hasta el momento en instalaciones de este tipo. El resultado de tal aplicación - curva criterio para el público - contribuyó, juntamente con el informe preliminar de seguridad y otros documentos de diseño, al otorgamiento de la licencia de construcción de dicha central nuclear. La norma AR 4.1.3. fue aplicada asimismo al reactor de investigación y producción RP 10, diseñado y construido por la CNEA en Perú. En este caso, la verificación del cumplimiento de los criterios radiológicos relativos a accidentes contribuyó a demostrar la factibilidad de licenciar la instalación en el marco normativo argentino, una de las cláusulas contractuales establecidas en el proyecto del reactor mencionado.

Cabe mencionar que se han desarrollado también en otros países - por ejemplo Reino Unido y Canadá - criterios de aceptación del riesgo radiológico (Referencia 4).

En diversas oportunidades sin embargo y en distintos foros, se efectuaron críticas a los criterios de aceptación del riesgo radiológico del sistema regulatorio argentino. Entre las de mayor relevancia cabe citar las siguientes:

a) el riesgo es individual y excluye por lo tanto la consideración del detrimento asociado a la dosis colectiva o de otras consecuencias derivadas de los accidentes.

b) los criterios no permiten optimizar el diseño en cuanto a la seguridad de una instalación; en otras palabras, no dan respuesta a la cuestión "*how safe is safe enough*?".

c) la verificación de cumplimiento de los criterios requiere la realización de un análisis probabilístico de seguridad (APS), el cual se considera técnicamente como "dificultoso" y poco confiable debido a las incertezas, especialmente las asociadas a la segunda etapa de este análisis.

En el ámbito internacional también se formularon críticas al APS en general, considerado como metodología para la determinación de la frecuencia de daños severos en el núcleo de reactores y de fallas en la contención de centrales nucleares de potencia. El INSAG hizo pública en este caso la defensa del APS (Referencia 8) y, por otra parte, caracterizó a esta herramienta como un medio importante para la prevención de accidentes y como complemento del juicio

ingenieril, de la experiencia operativa y de los conceptos de defensa en profundidad aplicados al diseño de reactores nucleares (Referencia 9).

A continuación se discuten los aspectos favorables y desfavorables de los criterios de aceptación del riesgo radiológico del sistema regulatorio argentino, y se evalúa la factibilidad, y la conveniencia o no de introducirles modificaciones que amplíen la base conceptual en la que estos criterios se fundamentan, teniendo en cuenta la experiencia local e internacional en la materia y las críticas mencionadas en los puntos a, b y c precedentes.

3. DISCUSIÓN

3.1. Aspectos favorables del sistema argentino de aceptación del riesgo radiológico

3.1.1) Los criterios de aceptación del riesgo radiológico del sistema regulatorio argentino son consistentes con el principio de limitación de dosis usado con fines de protección radiológica, recomendado por la CIPR. El objetivo de los criterios es limitar el riesgo individual asociado a exposiciones potenciales a valores del mismo orden de magnitud que el riesgo individual asociado a las exposiciones normales que ocurren corrientemente en las instalaciones o prácticas.

3.1.2) La experiencia argentina en la aplicación de las normas regulatorias sobre aceptación del riesgo radiológico es aceptable. En efecto, en el caso de reactores nucleares, la normativa se aplicó al diseño conceptual de la central nuclear Atucha II y se aplicará también al diseño definitivo de esta instalación como requisito para el otorgamiento de la licencia de operación (no fue aplicada a las centrales nucleares Atucha I y Embalse, por ser la normativa posterior al diseño de estas dos instalaciones). Se aplicó asimismo al reactor RP 10, por las razones expuestas en la sección 2.

3.1.3) La verificación del cumplimiento de los criterios de aceptación del riesgo radiológico es relativamente simple y de fácil comprensión. Un conjunto de duplas de valores de dosis y de probabilidad de ocurrencia de las secuencias accidentales que dan lugar a exposiciones a la radiación de personas, define puntos que, si se localizan en la zona aceptable del espacio de las figuras 1 o 2 (Anexo II), se considera que el diseño verifica los criterios de aceptación de riesgo. En caso contrario se deben realizar modificaciones del diseño de la instalación tales que, o se reducen las dosis en el grupo crítico o se reduce la probabilidad de ocurrencia de las secuencias accidentales correspondientes.

3.1.4) La necesidad de contar con un APS completo (nivel I, II y III) para verificar el cumplimiento de los criterios de aceptación del riesgo radiológico, no debe considerarse como una dificultad sino más bien como una ventaja. En efecto, se ha reconocido reiteradamente, no solo en el ámbito local sino en el internacional, la utilidad de los resultados de un APS como medio idóneo para detectar deficiencias de diseño o de procedimientos que, una vez corregidas, mejoran la seguridad (por ejemplo, la interconexión eléctrica entre las centrales nucleares Atucha I y Atucha II y la instalación de válvulas de aislación en la línea auxiliar del presurizador de la central nuclear Atucha I). Por otra parte, el perfeccionamiento continuo de las herramientas de cálculo y la creciente capacidad computacional, contribuyen a relativizar cada vez más las “dificultades técnicas” imputadas al nivel II del APS. Asimismo, el aporte continuo de información proveniente de la experiencia de instalaciones en operación permite disponer de bancos de datos de proceso y de fallas de componentes, equipos y sistemas robustos y actualizados, lo cual permite reducir progresivamente las incertidumbres asociadas a esta clase de análisis.

3.1.5) Acorde a lo establecido en la norma AR 10.1.1, los criterios de aceptación del riesgo radiológico deben aplicarse a instalaciones Clase I. La extensión de las aplicaciones de estos criterios a otras instalaciones - que no sean reactores nucleares (Referencia 3).- permitiría adquirir una experiencia valiosa en la materia.

3.2. Aspectos desfavorables del sistema argentino de aceptación del riesgo radiológico

3.2.1) La evaluación del diseño de una instalación basada en criterios de aceptación del riesgo radiológico no permite optimizar formalmente el diseño minimizando el riesgo; en rigor, solo posibilita la detección de aquellas secuencias accidentales cuyos valores asociados de probabilidad anual de ocurrencia y dosis efectiva en las personas determinan un punto localizado en la región no aceptable de las figuras 1 o 2 (Anexo II). Existe sin embargo la posibilidad de que futuros desarrollos permitan aplicar los criterios de aceptación del riesgo radiológico a la optimización del diseño de instalaciones nucleares. (Referencia 7).

3.2.2) Los criterios de aceptación del riesgo radiológico del sistema regulatorio argentino sirven como método para la toma de decisiones, basado solamente en el riesgo individual. Es pertinente entonces plantear la cuestión de porqué no incluir también a la población y a otros factores sociales, ambientales e incluso económicos cuando se trata de reducir riesgos. Este enfoque (método multiatributos) es mencionado en algunas publicaciones con el término de **riesgo social** (Referencia 5) y es utilizado para representar el impacto total de un accidente. La dosis individual, la dosis colectiva y los valores económicos asociados a evacuación de personas, control de alimentos contaminados y descontaminación de suelos son algunas de las componentes del riesgo social a considerar en caso de accidentes con consecuencias radiológicas. Si bien la Teoría de la Decisión dispone de herramientas matemáticas idóneas para el tratamiento de estos casos, los intentos de utilización de esta teoría en el campo nuclear no han prosperado, dada la diversidad de los escenarios accidentales y la complejidad de las instalaciones nucleares involucrados. Un modelo más simple, aunque incompleto, es aquel que toma en cuenta solo algunas de las componentes del riesgo social, por ejemplo la dosis colectiva en la población afectada por el accidente o el número de muertes, debidas ya sea a los efectos severos o a los estocásticos de la radiación. Estos modelos, sin embargo, solo han demostrado su utilidad en algunos estudios con fines de comparación entre distintos tipos de reactores nucleares o distintas clases de accidentes(Referencia 6), pero no han sido utilizados para establecer criterios de aceptación del riesgo. En suma, el desarrollo de un método para la toma de decisiones que tenga en cuenta el impacto total de los accidentes con consecuencias radiológicas - es decir basado en el enfoque del riesgo social - es técnicamente posible, aunque inabordable en la práctica. Modelos más simples que tienen en cuenta una sola componente del riesgo social han sido utilizados en algunos estudios comparativos, pero no fueron empleados hasta el momento para el desarrollo de criterios de aceptación de riesgos sociales.

3.2.3) Las dificultades técnicas que se presentaron al encararse la realización del APS nivel II de centrales nucleares condujeron a los especialistas a plantearse la posibilidad de simplificar el desarrollo de esta clase de estudios, y reemplazar el criterio de aceptación del riesgo en función de probabilidad de accidentes y magnitud de las consecuencias radiológicas, por otros indicadores más modestos. Por ejemplo, el Grupo Internacional Asesor en Seguridad Nuclear (INSAG) introdujo en uno de sus informes (Referencia 5), objetivos de riesgo ("risk targets") expresados mediante los siguientes valores de frecuencia de ocurrencia de eventos con daños severos al núcleo de reactores de potencia: a) para instalaciones existentes $< 10^{-4}$ reactor-año^{-1} y b) para futuras instalaciones $< 10^{-5}$ reactor-año^{-1} . Con el mismo enfoque pueden fijarse objetivos de confiabilidad de sistemas importantes para la seguridad, tales como los de extinción del reactor, refrigeración de emergencia del núcleo y cierre de la contención de una central nuclear. Estos objetivos podrían ser considerados como indicadores del valor aceptable del riesgo radiológico impuesto por la instalación a la cual están asociados; pero para ello deberían guardar algún tipo de relación con la magnitud de las consecuencias radiológicas de las exposiciones potenciales resultantes de accidentes en dicha instalación. Explicitar dicha relación ha probado ser, hasta el momento, una tarea problemática o, al menos, de una complejidad similar a la de un APS completo.

4. CONCLUSIONES

El sistema regulatorio argentino cuenta con criterios de aceptación del riesgo radiológico adecuados y consistentes con los principios de protección radiológica de las personas. Estos criterios han sido utilizados aceptablemente en reactores nucleares y no se prevén impedimentos para su aplicación a otras instalaciones Clase I.

Las críticas formuladas a estos criterios así como algunas dificultades técnicas asociadas a su utilización como herramienta para la toma de decisiones no los descalifican y no justifican por el momento la modificación de la base conceptual en la que los mismos se fundamentan.

Se debe continuar con la aplicación de los criterios de aceptación del riesgo radiológico según lo prescriben las normas regulatorias y asimismo extender su aplicación a otras instalaciones Clase I además de los reactores nucleares.

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ANEXO I

RIESGO RADIOLÓGICO EN EL CASO DE EXPOSICIONES POTENCIALES

El criterio de aceptación argentino se basa en la concepción probabilística del riesgo radiológico individual y emplea la filosofía subyacente del sistema de limitación de dosis usado con fines de protección radiológica, recomendado por la CIPR. El objetivo del criterio es limitar el riesgo individual asociado a exposiciones potenciales, a valores del mismo orden de magnitud que el riesgo individual asociado a las exposiciones normales que ocurren en las instalaciones o prácticas. Técnicamente el riesgo individual se define como la probabilidad de que, en un dado período, un individuo se exponga accidentalmente a radiaciones ionizantes y fallezca debido a tal exposición; su expresión matemática es la siguiente:

$$R = P(E) \bullet P(F/E)$$

Donde:

R: riesgo individual asociado a un accidente radiológico.

P(E): probabilidad de exposición dado el accidente.

P(F/E): probabilidad de fallecimiento atribuible a la exposición.

La probabilidad P(E) depende de diversos factores tales como las características tecnológicas de la instalación o práctica de que se trate, las características del emplazamiento (topográficas, meteorológicas, etc.) y la ubicación de las personas asociadas.

La probabilidad condicional P(F/E) ,en cambio, es función de la dosis D incurrida por el individuo expuesto; aumenta al principio de manera lineal en el dominio de dosis de hasta 1 Sv aproximadamente - con una pendiente $\alpha \cong 5 \times 10^{-2}$ Sv $^{-1}$ - y luego de manera sigmoidea para dosis mayores que 1 Sv hasta alcanzar valores cercanos a la unidad para dosis del orden de 6 Sv o mayores (Referencia 4).

La Autoridad Regulatoria ha limitado el riesgo radiológico individual en el caso del público, al valor $R = 10^{-6}$ por año (valor correspondiente al grupo crítico del público) y en el caso de los trabajadores, al valor $R = 10^{-5}$ por año (valor correspondiente al trabajador más expuesto).

ANEXO II

Para estimar el riesgo se identifica el conjunto de secuencias accidentales previsibles con implicancias radiológicas para miembros del público o para los trabajadores - según corresponda - y, posteriormente, se calculan las probabilidades anuales de ocurrencia de tales secuencias, así como las dosis efectivas en el grupo crítico del público o en el trabajador más expuesto - según sea el caso - resultantes de las respectivas exposiciones potenciales (Referencia 4).

El criterio establece que para el caso de accidentes con consecuencias radiológicas para el público, ninguna secuencia accidental debe tener una probabilidad anual de ocurrencia que, graficada en función de la dosis efectiva, resulte en un punto ubicado en la zona no aceptable de la "curva criterio para el público" (Figura 1); de igual manera establece que para el caso de accidentes con consecuencias radiológicas para los trabajadores, ninguna secuencia accidental debe tener una probabilidad anual de ocurrencia que, graficada en función de la dosis efectiva, resulte en un punto ubicado en la zona no aceptable de la "curva criterio para los trabajadores" (Figura 2).

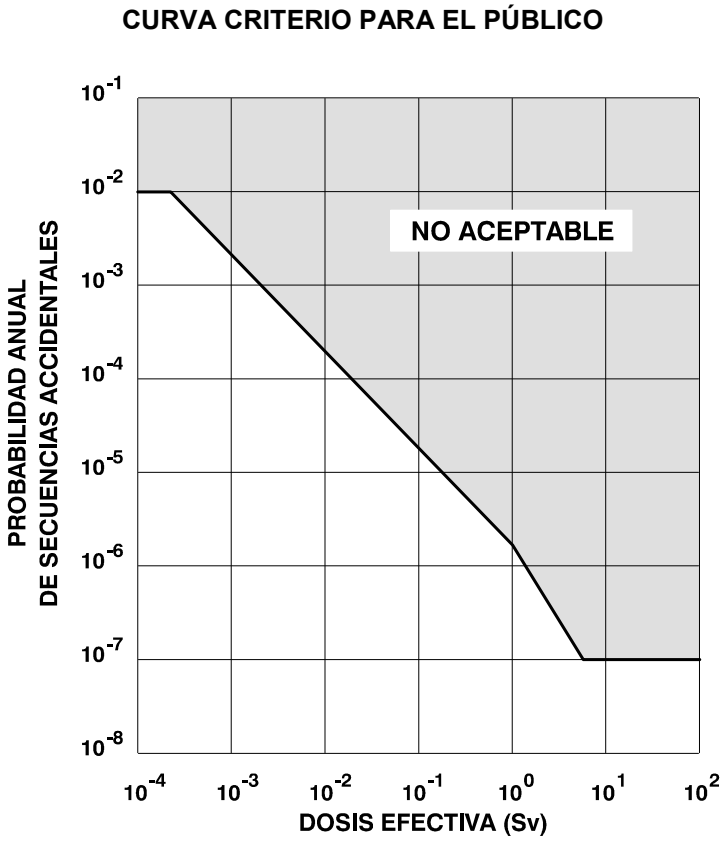

Figura 1

CURVA CRITERIO PARA LOS TRABAJADORES

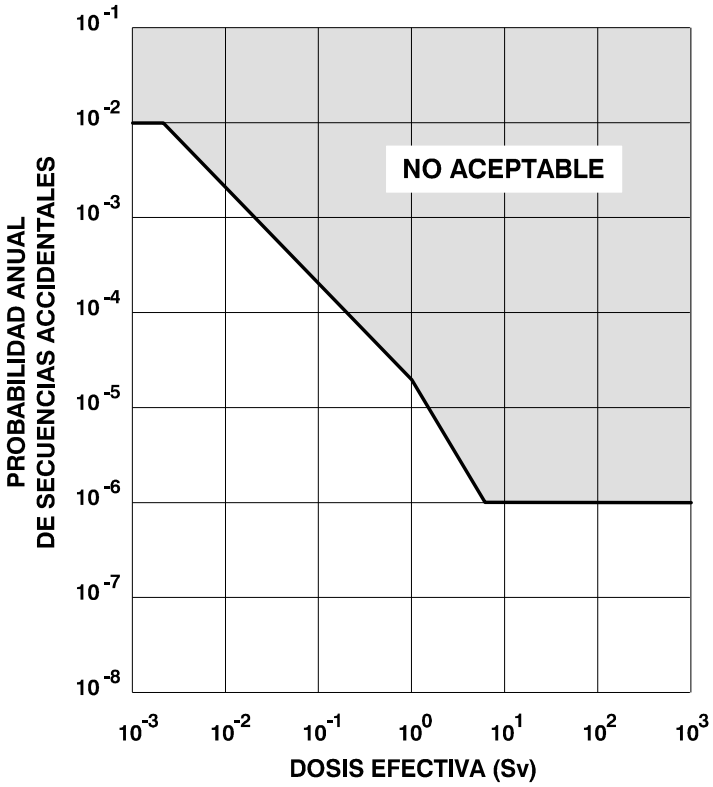

Figura 2

Informe sobre TENORM situación nacional e internacional

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Publicado como PI 2/05 de la Autoridad Regulatoria Nuclear

INFORME SOBRE TENORM
SITUACIÓN NACIONAL E INTERNACIONAL

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RESUMEN

En el presente informe se analiza la situación de los materiales TENORM en el mundo con el fin de delinear las acciones a seguir frente al tema en nuestro país. Se presentan los distintos criterios regulatorios adoptados y se analiza la situación nacional desde el punto de vista legal. El informe se enfoca fundamentalmente en la industria del gas y del petróleo, ya que de acuerdo a lo expuesto en un informe preliminar[1], esta industria sería la más significativa desde el punto de vista de su contenido en TENORM.

ABSTRACT

In the present report, the situation of TENORM materials around the world is analyzed in order to plan future steps regarding this subject in our country. The different regulatory criteria adopted are presented and the national situation is analyzed from the legal point of view. The report is fundamentally focused on petroleum and gas industry, due to, as it was exposed in a preliminary report[1], this industry would have the most important TENORM content.

CONCEPTOS BÁSICOS

Se conoce como NORM (naturally occurring radioactive material) a los materiales radiactivos de origen natural. Ciertos minerales contienen niveles significativos de radionucleidos naturales que son extraídos y procesados junto a otros elementos. El procesamiento posterior de estos materiales puede hacer aumentar la concentración de estos radionucleidos.

Algunas industrias en su procesamiento concentran estos radionucleidos naturales pudiendo causar algún riesgo a las personas si no son controladas. Estos materiales naturalmente radiactivos que son concentrados por ciertas industrias se llaman TENORM (technologically enhanced naturally occurring radioactive material). Es importante destacar que a pesar de la diferencia conceptual entre NORM y TENORM, muchas veces se utilizan el término NORM para referirse a TENORM.

Los TENORM se encuentran en algunas corrientes de residuos generados en industrias no nucleares; por ejemplo desechos de metales, barros, escorias, fluidos. Estos materiales, los productos derivados y los productos finales del procesamiento pueden aumentar la exposición tanto de los trabajadores como de los miembros del público. A pesar de que el contenido de los radionucleidos en estos materiales puede ser considerado relativamente pequeño, en algunos ambientes de trabajo pueden presentarse niveles importantes de radiación. Esto significa que trabajadores pertenecientes a industrias no nucleares pueden recibir una dosis de radiación significativa.[2] Estos lugares podrían requerir alguna forma de control regulatorio para proveer una adecuada protección a los trabajadores. La fuente más importante de radiactividad en los TENORM se debe a la presencia de los isótopos producto del decaimiento del uranio y el torio. Como ejemplo, en la industria del gas y el petróleo los radionucleidos más importantes a tener en cuenta son el ^{226}Ra de la serie de decaimiento del ^{238}U, y en menor grado el ^{228}Ra, de la serie de decaimiento del ^{232}Th. [3,4,5].

PRODUCCIÓN DE GAS Y PETRÓLEO

La existencia de materiales radiactivos de origen natural en formaciones geológicas de interés petrolífero está ampliamente documentada. La cantidad de material radiactivo es variable según las características de la corteza terrestre de la zona en cuestión. Los materiales conteniendo radionucleidos de origen natural encontrados en los campos petroleros se hallan en las formaciones subterráneas, o reservorios de gas y de petróleo, originados en la era Jurásica. Los isótopos radiactivos naturales contenidos en las rocas reservorio pertenecen a las series naturales del ^{238}U y del ^{232}Th. Los isótopos de estas dos cadenas existen en la naturaleza en las rocas sedimentarias que almacenan el petróleo, adonde migraron desde las rocas madre a partir de su disgregación química y la erosión. Tanto en el caso de la industria del petróleo como en la del gas, las técnicas de forzado de la salida a la superficie del material crudo incluyen la recirculación de agua que luego es extraída junto con el petróleo y el gas.

Los materiales NORM son típicamente transportados a la superficie junto con el agua salada. Se producen cambios de presión y descenso de la temperatura, lo que provoca la precipitación de sulfatos y carbonatos en las tuberías y en las superficies internas de los equipos. La similitud química del radio con el bario lleva a la coprecipitación selectiva en barros e incrustaciones de los isótopos del radio. Las incrustaciones y los barros resultantes, en general contienen radio y otros productos de la cadena de decaimiento del uranio y del torio. El material radiactivo natural que no forma incrustaciones, aparece en las piletas de recepción de agua de purga. Otros radionucleidos de interés, particularmente en los equipos de gas, son el gas radón y el ^{210}Pb, que suele formar una fina capa sobre la superficie interna de los equipos de procesamiento[6, 7].

Los principales aspectos de protección radiológica relacionados con las incrustaciones son la irradiación gamma del personal y la contaminación interna originada en la remoción de las mismas. La concentración de actividad reportada va desde 1 a 1000 Bq/g de ^{226}Ra, con valores algo menores de ^{228}Ra (dependiendo de la antigüedad de la incrustación). La concentración de actividad es fuertemente dependiente de las variaciones de presión y temperatura. Una característica es que las concentraciones superiores a 100 Bq/g existen sólo en pequeñas cantidades de material (kg), y en grandes cantidades (toneladas) en concentraciones de 1-10 Bq/g.

SITUACIÓN INTERNACIONAL

Debido a las importantes dosis que pueden recibir los trabajadores de industrias no nucleares a causa de los materiales NORM y TENORM, y las concentraciones de actividad que poseen los residuos de estas industrias, la preocupación a nivel mundial por este tema se ha ido incrementando en los últimos años y se ha convertido en un tema de debate. Luego de más de una década de mediciones y evaluaciones dosimétricas, los distintos países comenzaron a delinear diferentes posturas regulatorias frente al tema. Aun en la actualidad existen diferencias de criterio que están siendo discutidas en el ámbito internacional.

Las autoridades regulatorias de muchos países prescribieron niveles de exención para concentraciones de actividad, actividad total y tasas de dosis para actividades que no requieren licenciamiento, utilizando como base las Normas Básicas Internacionales de Seguridad para la Protección contra la Radiación Ionizante y para la Seguridad de las Fuentes de Radiación (BSS-115)[8]. En una primera evaluación en distintas industrias, tales como la industria del fosfato, estaño y gas y petróleo, se vio que los valores variaban ampliamente, y que un importante número de mediciones de concentraciones de actividad se encuentra muy por encima de los niveles de exención de 1 y 10 Bq/g para ^{232}Th y ^{226}Ra respectivamente. Las tasas de dosis externas en la minería del estaño y en la industria del gas y el petróleo mostraron niveles de 1,1 a 5,2 $mSv.a^{-1}$ y 1 a 40 $mSv.a^{-1}$ respectivamente.[9]

Estos datos demuestran que estas industrias generan dosis importantes, nada despreciables, superando los niveles de exención y aun los límites de dosis establecidos para la industria nuclear.

a) IAEA

Dentro de la normativa internacional existente, en las BSS -115[8], es importante destacar dos párrafos referidos a la exposición a fuentes naturales: un párrafo dentro de la sección 2.1, donde se enumeran las prácticas a las que deberán aplicarse las Normas:

"*Las prácticas a las que deberán aplicarse las Normas son, en particular: ....c) las prácticas que conlleven exposición a fuentes naturales que, según especifique la autoridad reguladora, requieran control;*"...

El otro párrafo, en la sección 1.4, da como ejemplo de exclusión:

"la concentración, no modificada, de los radionucleidos presentes en la mayor parte de las materias primas".

La referencia a "la mayor parte de las materias primas" implica que pueden perfectamente existir algunas industrias que usan los NORM, donde las concentraciones de radiactividad sean bastante elevadas como para justificar que sean objeto de consideración y control. La referencia a "las concentraciones no modificadas" señala que la elaboración de algunas materias primas, que puede tener concentraciones de radiactividad relativamente normales, puede generar productos o deshechos con niveles mucho más altos.

Para la liberación de material proveniente de la industria nuclear, la IAEA adopta 10 $\mu Sv.a^{-1}$ como criterio de dosis individual por práctica, y está considerando establecer una optimización en cada caso individual de regulación de TENORM[10]. Esto significaría la liberación de grandes cantidades de material TENORM provenientes de industrias no nucleares a niveles mucho más altos que el valor criterio adoptado para la industria nuclear. En el 2001, la IAEA sugirió: que se podría tomar un solo set de valores de niveles de exención, clearance y commodities en términos de Bq/g, aplicable a todos los materiales excepto agua y comestibles; que la BSS fuese modificada introduciendo una definición de su alcance, y que se remuevan los niveles existentes de exención y clearance. Esta propuesta se encuentra en el reporte titulado: The Scope of Radiation Protection Safety Standards Strategy for Rationalisation of Policy [10]. En julio del 2001 se reunió un comité técnico y se discutió la propuesta de 300 $\mu Sv.a^{-1}$ para situaciones "de facto" (TENORM) y 10 $\mu Sv.a^{-1}$ para prácticas (industria nuclear) [10]. Todo esto sugiere que la situación es bastante confusa. En resumen, la Comunidad Europea y la IAEA proponen dos estándares distintos, un criterio de dosis individual de 10 $\mu Sv.a^{-1}$ para liberar materiales de la industria nuclear, y un criterio de 300 $\mu Sv.a^{-1}$ para material de las industrias no nucleares. Esto solo logra complicar los esfuerzos para lograr consistencia, armonización y el movimiento de los materiales más allá de las fronteras. También significaría que la radiactividad de la esfera nuclear y de las industrias no nucleares son juzgadas de diferente manera, teniendo exigentes condiciones para liberar material de las industrias nucleares y permitiendo treinta veces mas para cantidades mucho mayores de radiactividad de las industrias no nucleares. Al hacer esto, el mensaje al público es que la radiactividad nuclear es treinta veces más peligrosa que la radiactividad de los TENORM. El mismo nucleido, en la misma concentración, puede tanto ser dispuesto en un reservóreo profundo como liberado para su uso en la reparación de caminos, dependiendo si su origen es nuclear o no. Con respecto a esto, el US National Academy of Sciences, ha claramente refutado toda posible razón, desde el punto de vista de la protección radiológica, para tratar al material radiactivo proveniente de la industria nuclear y al que proviene de las industrias NORM no nucleares con diferentes estándares de evaluación de riesgo, ya que la dosis absorbida en los tejidos depende solamente del tipo de radiación y su energía y no de la fuente de radiación [10, 11]. De todas formas, la adopción del criterio de exención dado por la BSS para la industria nuclear no es compatible con las características inherentes a las industrias no nucleares. Los criterios de exención y clearence no deberían ser aplicados textualmente ya que, no solo se verían recargadas las industrias en cuestión, sino también las autoridades regulatorias y la economía nacional, sin un aparente beneficio desde el punto de vista de la protección radiológica[12].

Con respecto a los desechos radiactivos, en un Boletín del IAEA[13] escrito entre otros por el Ing. A. González, se analiza el caso de los residuos originados por las industrias que procesan NORM. Las regulaciones relativas a los desechos radiactivos se han concentrado en las prácti-

cas que hacen uso de las fuentes artificiales de radiactividad. Sin embargo, existe otra esfera donde se podrían exponer argumentos en contra y a favor de la necesidad de intervenciones de regulación aplicando los criterios de la protección radiológica: los desechos de las industrias que utilizan NORM. Las recomendaciones del ICRP confirmaron la idea de que, en principio, esas industrias pueden ser candidatas a regulación. Estas industrias pueden producir desechos radiactivos que contienen niveles mucho más altos de radiactividad que los niveles de exención. Estas situaciones son diferentes a las relacionadas con los radionucleidos artificiales en que se ha utilizado el concepto de trivialidad para decidir el alcance de la intervención de la regulación. Uno de los enfoques sería excluir esas industrias de las regulaciones, a menos que los niveles de actividad de los materiales utilizados fueran de tal magnitud que las dosis que estuvieran recibiendo fueran suficientemente altas como para causar preocupación. Otro enfoque que se basa en la decisión de que las industrias especificadas deben estar sujetas a regulación, es decir, constituyen una práctica en el contexto de la BSS. En tales casos, prever la exención de los requisitos de regulación puede ser útil, pero sería necesario definir las condiciones de dicha exención. Desde una perspectiva teórica podría interpretarse que se aplican diferentes "normas" en situaciones que incluyen radionucleidos artificiales y radionucleidos provenientes de materiales TENORM. Por esa razón, algunos han propuesto que las industrias que procesan NORM deberían ser reguladas de la misma forma que las industrias relacionadas con la energía nuclear. Ello significaría que para la mayoría de los desechos, no sería adecuada la exención porque la exposición a las radiaciones no es trivial. El nivel de regulación variaría según los riesgos potenciales para los trabajadores y para el público, y en las industrias donde los riesgos causados por la radiación son bajos y la fuente o práctica es intrínsecamente segura, puede bastar una notificación del operador o propietario al órgano regulador en que comunique la existencia de la práctica y sus desechos.

b) Comunidad Europea

El COUNCIL EURATOM 96[14] en su alcance dice que, de acuerdo con TITLE VII, las directivas de este documento deben aplicarse no solo a las prácticas sino también a las actividades de trabajo (work activities) que involucren la presencia de fuentes naturales de radiación y lleven a un aumento significativo de la exposición de los trabajadores o miembros del público que no puede dejar de considerarse desde el punto de vista de la protección radiológica. TITLE VII dice que cada Estado Miembro debe asegurar la identificación por medio de inspecciones, o cualquier otro medio apropiado, de aquellas work activities que puedan ser de interés. Para cada work activity declarada de interés, el Estado Miembro requerirá monitorear la exposición y, donde sea necesario, implementar medidas correctivas para reducir la exposición. Esas medidas correctivas pueden estar en conformidad con TITLE X (intervenciones), o bien estar de acuerdo con las secciones referidas a prácticas en forma total o parcial.

El Radiation Protection 122 (parte II)[15] da guía a los Estados Miembro sobre como podrían aplicarse los conceptos de reporte, autorización, niveles de exención y clearance a las work activities, tal como se aplica a las prácticas (industria nuclear). Se define como work activity aquella actividad donde la presencia de fuentes de radiación natural lleven a un incremento significativo de la exposición de trabajadores y miembros del público (el material no es usado por sus propiedades físiles o radiactivas). En su sección correspondiente a Exención y Clearance, se aclara que los niveles de exención referidos en el artículo 3 de la EC-BSS fueron derivados para el uso de radionucleidos artificiales en escala moderada y a priori no son aplicables a radionucleidos de ocurrencia natural. Como las cantidades de material a considerar en estos casos son de gran volumen, los conceptos de exención y clearance convergen, y por lo tanto es apropiado que se utilicen los mismos niveles de referencia para ambos conceptos. La definición de esos valores para fuentes naturales no puede provenir del criterio de riesgo trivial establecido en el Anexo I de la EC-BSS: para work activities, la exposiciones anuales individuales pueden ser mucho mayores que 10 $\mu Sv.a^{-1}$ y las dosis colectivas pueden ser muy importantes. Los niveles de exención-clearance para radionucleidos de ocurrencia natural deberían ser establecidos a un nivel de dosis mayor que el utilizado para prácticas. El artículo 31 del Tratado Euratom propone para work activities, establecer un incremento de dosis anual efectiva de 300 $\mu Sv.a^{-1}$ como criterio para exención y clearance. Esto puede ser expresado en términos de concentración de actividad planteando los escenarios. La elección de este criterio está justificada por lo siguiente: es comparable o aún menor que las variaciones regionales en dosis total efectiva de radiación de fondo; es coherente con el nivel de exención propuesto para

materiales de construcción[16]; es coherente con la restricción de dosis que se utiliza para el control de efluentes (ICRP recomienda 300 μSv.a^{-1} para prácticas, pudiendo ser apropiado un valor mayor de hasta 1 mSv.a^{-1} para work activities) y por último, es menor que el punto de referencia más bajo propuesto para el control de la exposición de trabajadores en work activities.

Con respecto al control de la exposición de trabajadores, la Comunidad Europea llevó a cabo un proyecto para enfrentar el problema estableciendo niveles de referencia para el control regulatorio de las industrias involucradas. Este estudio se llevó a cabo junto con el NRBP (National Radiological Protection Board) y el CEPN (Centre d'Etude sur l'Evaluation de la Protection dans le domaine Nucléaire)[3]. Se elaboraron los documentos Radiation Protection 95 y Radiation Protection 107. Estos documentos son guías para asistir a los Estados Miembros de la Comunidad Europea para poder implementar TITLE VII. Ofrecen una técnica simple de screening y categoriza a las industrias según el criterio de dosis de radiación. Cuando la categorización sugiere una dosis significativa potencial, se debe hacer una caracterización específica del lugar. En el Radiation Protection 107 se describe en forma detallada los escenarios y parámetros de exposición que derivaron en los niveles de referencia propuestos en el documento anterior, Radiation Protection 95. Con el fin de establecer los niveles de referencia, se trabajó de la siguiente forma: primeramente se revisaron las industrias que contienen niveles aumentados de radionucleidos naturales con el fin de saber el rango de concentración de actividad de los radionucleidos involucrados en cada proceso. Luego, se establecieron los escenarios de exposición (que reflejan las condiciones de trabajo y prácticas en cada industria) para calcular las dosis involucradas y los niveles derivados. Finalmente, sobre la base de los resultados obtenidos, se propuso un sistema de clasificación con fines regulatorios explicando su filosofía y aplicación. Este sistema de clasificación se basa en lo siguiente: si la dosis es menor a 1 mSv.a^{-1}, entonces no se requieren precauciones especiales. Si las dosis exceden 1 mSv.a^{-1}, entonces puede ser aplicado un esquema para controlar la exposición. Si las dosis exceden 6 mSv.a^{-1}, puede ser apropiado definir un área controlada. Si la dosis efectiva es mayor a 20 mSv.a^{-1}, la práctica es claramente inaceptable ya que implica riesgos inaceptables.

c) EE. UU.

La US-EPA (Environmental Protection Agency), desde mitad de los años 70, comenzó a conducir estudios para determinar el riesgo a la salud humana y al medio ambiente de las liberaciones de TENORM provenientes de diversas industrias. La EPA se interesa por los TENORM por tres razones: en primer lugar, estos elementos tienen el potencial de causar niveles elevados de exposición a la radiación. Segundo, las personas no tienen conocimiento de estos materiales y necesitan información. En tercer lugar, las industrias que generan estos materiales pueden necesitar una guía adicional para el manejo y disposición de estos materiales de manera de proteger a las personas y al medio ambiente. La EPA ha comenzado a re-evaluar los riesgos de los materiales TENORM de algunas industrias. La estrategia se basa en los siguientes cuatro objetivos para abordar el problema[17]:

-Estudiar las industrias que generan materiales TENORM para determinar la composición de los residuos originados y el riesgo asociado.

-Identificar y estudiar los sitios que contienen materiales TENORM para tener una visión completa de la situación nacional del problema –la ubicación de los residuos, la composición de los mismos y el riesgo asociado.

-Desarrollar y proveer una guía para controlar las exposiciones debido a los residuos que contienen TENORM en forma segura y práctica.

-Trabajar junto a otras organizaciones que estén tratando el problema de TENORM, incluyendo agencias estatales y federales, grupos ambientales e industriales y organizaciones internacionales.

En los EE. UU. no existen regulaciones federales específicas para fuentes difusas de TENORM. La autoridad que poseen los Estados para regular estos asuntos deriva directamente de la Constitución, la cual atribuye a los Estados la responsabilidad primaria por la salud y la seguridad del público. Muchos Estados consideran que los TENORM quedan regidos por sus

regulaciones generales sobre radiación, en cambio otros creen que deberían tener regulaciones específicas. En la actualidad 13 estados poseen regulaciones sobre TENORM, y otros 9 están estudiando el tema. Las regulaciones que los distintos Estados adoptaron sobre estos materiales, se basan en las regulaciones designadas para mitigar el gas radón de las colas de estériles de mineral de uranio, así como la disposición de las mismas[18].

Louisiana fue el primer estado que desarrolló e implementó un programa regulatorio para NORM (Louisiana Department of Environmental Quality 1992)[19]. Bajo este programa, los operadores que manipulen o posean NORM por encima de ciertos valores deben tener una licencia de operación. Estas regulaciones alcanzan a todo material o equipamiento o terreno que haya sido contaminado con material NORM.

Todo material o deshecho está exento de regulación sólo si presenta, como máximo, las siguientes concentraciones:

-5 pCi/g (0,2 Bq/g) de ^{226}Ra ó ^{228}Ra por encima del fondo natural,

-150 pCi/g (5,5 Bq/g) por encima del fondo natural de otros radionucleidos,

-la tasa de exposición externa en todo lugar no debe exceder los 50 μR/h (0,5 μSv/h)

Los suelos están exentos de todo requerimiento sólo si, en un muestreo promedio de 100 m^2, las concentraciones son menores que los límites especificados a continuación:

-5 pCi/g (0,2 Bq/g) de ^{226}Ra o ^{228}Ra por encima del fondo natural en un promedio de los primeros 15 cm de suelo,

-15 pCi/g (0,55 Bq/g) en una profundidad superior a los 15 cm,

-30 pCi/g (1,1 Bq/g) de ^{226}Ra o ^{228}Ra en suelos a los 15 cm de profundidad si la dosis equivalente efectiva total a miembros el público no excede 1 $mSv.a^{-1}$.

En otras secciones se refiere al equipamiento de muestreo, la frecuencia de los mismos; en los casos no exceptuados existe una “Licencia General” que describe todos los requisitos para extraer, recibir, poseer, usar, almacenar y transferir NORM. También se describen los requerimientos para la protección de los trabajadores durante las operaciones.

En el caso de la regulación de Texas[20], se presentan las siguientes exenciones:

a) personas que reciben, poseen, usan, transfieren, transportan, almacenan, o distribuyen comercialmente:

1) Todo material o deshecho de la industria del gas y del petróleo está exento de regulación sólo si presenta, como máximo, las siguientes concentraciones:

-30 pCi/g (1,1 Bq/g) de ^{226}Ra ó ^{228}Ra por encima del fondo natural.

En suelo, en un muestreo promedio de 100 m^2, si las concentraciones son menores que los límites especificados a continuación:

-30 pCi/g (1,1 Bq/g) de ^{226}Ra ó ^{228}Ra por encima del fondo natural en un promedio de los primeros 15 cm de suelo,

-150 pCi/g (5,5 Bq/g) de otros radionucleidos.

2) Desechos que no provienen de la industria del gas y del petróleo:

-30 pCi/g (1,1 Bq/g) de ^{226}Ra ó ^{228}Ra en los primeros 15 cm de suelo siempre que la tasa de emanación de radón sea menor que 20 pCi/m^2s (0,74 Bq/m^2s),

-5 pCi/g (0,2 Bq/g) de ^{226}Ra o ^{228}Ra por encima del fondo natural en suelo, en un muestreo promedio de 100 m^2, sobre los primeros 15 cm de suelo donde la tasa de emanación de radón sea igual o mayor que 20 pCi/m^2s (0,74 Bq/ m^2s).

-150 pCi/g (5,5 Bq/g) de otros radionucleidos en suelo, en un muestreo promedio de 100 m^2, sobre los primeros 15 cm de suelo donde la tasa de emanación de radón sea menor que 20 pCi/m^2s (0,74 Bq/ m^2s)

b) todo material o equipamiento en el proceso de reciclado; tuberías o equipamiento de superficie usado en la producción de petróleo está exento si lo niveles de exposición no superan los 50 μR/h (0,5 μSv/h)

Para los casos no exceptuados existe una "Licencia General" que describe todos los requisitos a seguir en cada caso.

d) Canadá

Guías Canadienses para el uso de NORM[21]

Las guías se basan en las regulaciones recomendados por el ICRP y por la Comisión Canadiense de Seguridad Nuclear (CNSC). El principio básico de estas guías es considerar que los trabajadores y los miembros del público están expuestos a fuentes adicionales de radiación debido a actividades que están involucradas con materiales NORM y sostienen que los mismos principios de protección a la radiación que se aplican a las actividades nucleares deberían aplicarse a las actividades NORM. No hay distinción en cuanto al origen de la radiación y debería aplicarse el principio ALARA, en que si las dosis pueden ser reducidas por acciones razonables, aquellas acciones deberían tomarse, teniendo en cuenta factores sociales y económicos. El ICRP reconoce que todas las personas están sujetas a cierta exposición a la radiación de fondo. Sin embargo, aunque las dosis de prácticas ocupacionales sean tan bajas, cercanas al fondo, la práctica es injustificable si no hay un beneficio asociado. La pregunta a responder es: a qué incremento de dosis deberían comenzar a implementarse acciones de protección radiológica a los materiales NORM. El Federal Provincial Territorial Radiation Protection Commitee recomienda que la dosis anual efectiva *incrementada* para personas expuestas a NORM es: 20 mSv para los trabajadores que están expuestos a NORM como resultado de sus tareas regulares y 1 mSv para miembro del público y trabajadores incidentalmente expuestos, es decir que no trabajan regularmente con materiales NORM. Este incremento de dosis no incluye el fondo de radiación natural, excluyendo también el radón. Existe una *restricción de dosis*, que es un límite superior de dosis anual efectiva establecido para cada práctica, para miembros del público y trabajadores incidentalmente expuestos. Este valor se establece para que no supere el límite de dosis de 1 $mSv.a^{-1}$ considerando la exposición a todas las prácticas. El ICRP sugiere un valor de 0,3 $mSv.a^{-1}$. Estas guías adoptaron este valor como niveles de investigación. Si este valor es excedido, se debe hacer un control específico del lugar, realizando una evaluación dosimétrica. Las personas involucradas en la evaluación deben ser expertos en protección radiológica.

Programa de manejo de NORM

Las clasificaciones de este programa resumen los requerimientos para el manejo de materiales NORM. Estas son establecidas en base a la dosis máxima anual recibida tanto para miembros del público como para el trabajador.

La clasificación sería la siguiente:

-Cuando los incrementos de dosis anual efectiva son <0,3 $mSv.a^{-1}$ para el público y <1 $mSv.a^{-1}$ para el trabajador, las tareas involucradas son irrestrictas.

-Cuando se superan los 0,3 $mSv.a^{-1}$, se denomina NORM Management. El acceso al público debe ser restringido, se debe verificar que no se introduzcan cambios en las condiciones de trabajo.

-Cuando se superan 1 $mSv.a^{-1}$, se denomina Dose Management. El programa incluye: notificación a los trabajadores de las fuentes de radiación, consideración de los procedimientos de trabajo y ropa protectora para limitar la dosis, entrenamiento a los trabajadores para controlar y reducir dosis, se deben estimar las dosis de los trabajadores y las dosis deben ser reportadas al Registro Nacional de Dosis.

-Cuando se superan los 5 $mSv.a^{-1}$, se denomina Radiation Protection Management. Además de los requerimientos del punto anterior debe realizarse: un programa de protección radiológica similar al que se aplica a los trabajadores de la industria nuclear que exceden 5 $mSv.a^{-1}$; debe utilizarse equipamiento protectivo, ropa y procedimientos de trabajo para reducir las dosis a los trabajadores y debe asegurarse que ningún trabajador supere el límite de dosis ocupacional promedio de 20 $mSv.a^{-1}$ en cinco años.

Una vez que se haya implementado un programa de NORM, ya sea NORM Management, Dose Management o el Radiation Protection Management, se necesitan realizar revisiones periódicas, con el fin de verificar que no se hayan introducido cambios en el sistema que puedan afectar a las dosis de radiación. En cada caso se determinará la frecuencia de inspección en base a los cambios introducidos en cada lugar de trabajo. Debe siempre aplicarse el principio ALARA.

Límites derivados de trabajo para NORM

Estos límites derivados permiten estimar la dosis anual a través de cantidades que son medidas directamente en el lugar de trabajo. Luego se comparan los valores medidos en el lugar con estos límites derivados.

a) Tasa de exposición

Nivel derivado de investigación gamma: la tasa de dosis ocupacional que de un incremento gamma de 0,3 $mSv.a^{-1}$ es de 0,15 $\mu Sv.h^{-1}$.

Nivel derivado de Dose Management: la tasa de dosis ocupacional que de un incremento gamma de 1,0 $mSv.a^{-1}$ es de 0,5 $\mu Sv.h^{-1}$.

b) Concentración de radón

Como la concentración de este gas originado por una práctica no puede distinguirse de la concentración de background, los límites de dosis están basados en dosis total, no en incremento de dosis por una práctica. Se recomienda que la concentración de radón medida corresponda a una dosis menor que 5 $mSv.a^{-1}$.

Límite derivado de investigación: la concentración de radón que se toma como límite derivado es de 150 $Bq.m^{-3}$, porque se tiene en cuenta el background.

NORM Management: cuando la concentración de radón está entre 150 y 800 $Bq.m^{-3}$, se debe verificar toda la práctica y aplicar el criterio ALARA si es posible.

Radiation Protection Management: el valor derivado para radón es de 800 $Bq.m^{-3}$, que representa una dosis anual efectiva de 5 $mSv.a^{-1}$, considerando 2000 horas de trabajo al año y un factor de equilibrio de 0,4. Si se supera este valor, deben hacerse todos los esfuerzos posibles para bajarlo. El valor máximo sería 3000 $Bq.m^{-3}$ que significaría una dosis de 20 $mSv.a^{-1}$.

c) Límite anual de incorporación (ALI)

El límite anual de incorporación es la cantidad de material radiactivo que un trabajador puede ingerir o inhalar cada año resultando en una dosis anual efectiva de 20 $mSv.a^{-1}$. Si la ingesta excede un 25% del ALI (equivalente a 5 $mSv.a^{-1}$) se debe implementar el programa de protección radiológica que implica protección respiratoria y/ o limitación del tiempo de acceso.

Manejo de material NORM

Las Guías recomiendan que el material conteniendo NORM puede ser reciclado sin restricciones radiológicas cuando la dosis asociada no es mayor a 0.3 $mSv.a^{-1}$. Estos valores se denominan límites derivados de liberación incondicional.

El valor para ^{226}Ra es de 0,3 $Bq.g^{-1}$ para materiales *difusos* (grandes volúmenes de material con distribución homogénea de radiactividad y baja concentración). Cuando hay más de un radionucleido presente en una muestra, la suma de las relaciones entre la actividad medida de cada radionucleido y su correspondiente limite derivado de liberación no debe exceder 1. Para materiales *concretos* (pequeños volúmenes de material y concentración de radionucleidos importante) el límite derivado incondicional es de 10^4 Bq totales.

Para contaminación superficial sobre equipos y herramientas, la liberación incondicional se realiza hasta un máximo de dosis anual efectiva de 0,3 $mSv.a^{-1}$ sobre el personal. Para contaminación superficial con material NORM tipo discreto el límite derivado incondicional a 50 cm es de 0,5 $\mu Sv.h^{-1}$ y 1 $Bq.cm^{-2}$ en un área de 100 cm^2.

Situación nacional

En nuestro país, no existe una regulación para los materiales TENORM. En lo concerniente a las radiaciones ionizantes, la Norma Básica de Seguridad Radiológica (NBSR) establecida por la ARN[22] provee un sistema de control regulatorio para la protección de los trabajadores y miembros del público de fuentes de radiación ionizante originados en la industria nuclear.

Sin embargo, al analizar en detalle la NBSR, se encuentra lo siguiente:

En la revisión 3 de esta Norma se establece en su alcance:

"Quedan exentos de esta norma y del control regulatorio, siempre que la Autoridad Regulatoria no entienda lo contrario, los siguientes casos:

- Toda utilización de materiales radiactivos naturales a los cuales no se les haya incrementado, tecnológicamente, la actividad por unidad de masa."

En la sección referente a Intervenciones, presenta situaciones de intervención que son aplicables más allá del ámbito de la actividad nuclear:

"La presente norma se aplica en los siguientes casos de intervención:

Situaciones crónicas de exposición a ciertas fuentes naturales de radiación que así lo requieran.

.....

Cualquier otra situación de intervención así considerada por la Autoridad Regulatoria."

Asimismo, sugiere un nivel de intervención para la concentración de radón en aire en viviendas:

"Cuando la concentración promedio anual de radón en el interior de viviendas exceda 400 becquerel por metro cúbico se deben adoptar soluciones de ingeniería para ventilar los ambientes y reducir la emanación del gas."

Además, en el artículo 8 de la Ley 24804, se establecen los fines de las funciones de la Autoridad Regulatoria

"ARTÍCULO 8°.- La Autoridad Regulatoria Nuclear deberá desarrollar las funciones de regulación y control que le atribuye esta ley con los siguientes fines:

a)Proteger a las personas contra los efectos nocivos de las radiaciones ionizantes..."

Con respecto a lo enunciado por la NBSR Rev. 3, los materiales TENORM no quedarían exentos ya que aquellas situaciones donde a los materiales naturales se les haya incrementado tecnológicamente la actividad por unidad de masa no están contemplados en la exención. Por otro lado, los materiales NORM estarían exentos siempre que la Autoridad Regulatoria no entienda lo contrario.

El inciso a) del artículo 8 de la Ley Nuclear permitiría a la ARN ampliar su ámbito de acción a toda situación de exposición dentro ó fuera de la actividad nuclear.

Es importante destacar que la ARN, en concordancia con el ICRP 60, adoptó en nuestro país un nivel de intervención para la exposición a gas radón en viviendas (radiación de origen natural), debido a la magnitud de las dosis individual y colectiva provocadas por el gas. El uso del nivel de acción para iniciar una intervención no es mandatoria. El nivel de acción de la concentración de radón adoptado corresponde a un valor de 400 $Bq.m^{-3}$, considerando un factor ocupacional de 7000 horas y un factor de equilibrio de 0,4. Esta exposición promedio continua implicaría una dosis anual efectiva cercana a 6 $mSv.a^{-1}$ (un equivalente a los $^{3}/_{10}$ del límite de dosis de los trabajadores). Esto constituye un precedente dentro de este órgano regulador en plantear una recomendación referida a radiación de origen natural.

Para comenzar a evaluar la posibilidad de regular aquellas industrias no nucleares que concentren NORM, primero se debería tener un claro panorama de la situación en el país. Para ello es necesario realizar una correcta evaluación de los riesgos asociados a los trabajadores pertenecientes a estas industrias, contando con una adecuada caracterización de la distribución de los materiales TENORM, un número de mediciones estadísticamente significativo, analizando también los escenarios en los cuales se desarrollan estas tareas. La información con la que se cuenta en la actualidad en nuestro país dista mucho de estas premisas, ya que los datos existentes son escasos y existe poca información sobre los escenarios de trabajo reales. Algunos autores adoptan como niveles de referencia valores equivocados y obsoletos, sacando conclusiones apresuradas con pocos datos, sin ningún estudio estadístico serio[23- 31].

Finalmente, para cumplir con el objetivo de tener un relevamiento completo de la situación nacional, habría que, no sólo elaborar un plan de mediciones acordado entre la ARN, otras organizaciones gubernamentales y las empresas en cuestión, sino también aprovechar aquellas situaciones donde las empresas piden asesoramiento sobre este tema para evaluar la seguridad radiológica de sus trabajadores.

CONCLUSIONES

- Para comenzar a estudiar las alternativas regulatorias aplicables a las industrias no nucleares que concentran radionucleidos naturales se debe, en primer lugar, realizar una correcta evaluación de la situación nacional: conocer la distribución de estos materiales en los distintos escenarios, realizar un número de mediciones estadísticamente significativo, y caracterizar los escenarios en los cuales se desarrollan estas tareas. Una vez que se obtenga un panorama claro se podrá decidir si dadas las condiciones existentes se requiere implementar medidas regulatorias o no. En caso de considerarse necesario la aplicación de algún tipo de regulación, en la elaboración de la misma debe tenerse en cuenta el impacto socioeconómico que esta producirá.

- En caso de que no se decida adoptar medidas mandatorias, sería muy importante que la ARN, ya sea en forma individual o colaborando con otras organizaciones, elabore guías y recomendaciones para el manejo seguro de estos materiales.

- En el presente informe se describen, a modo de ejemplo, las iniciativas regulatorias de varios países que pueden servir como guía u orientación. Cabe destacar que existen inconsistencias entre los distintos países con respecto a los criterios de dosis a partir de los cuales se originan los niveles de clearance y exención para las industrias no nucleares. El concepto de trivialidad no puede ser aplicado en estos casos. Se plantea entonces la discusión acerca de si estas industrias deben ser sometidas a los mismos niveles de control regulatorio que los de la industria nuclear, o ser tratados en forma diferente. Mientras que la primer opción requiere una gran infraestructura para permitir a la autoridad regulatoria investigar todas las prácticas que podrían verse involucradas, la segunda opción implica que la exposición a la radiación de los TENORM conlleva un riesgo menor que el de la industria nuclear. Al analizar el tema regulatorio, la ARN debería evaluar estas posturas.

- La ARN no ha establecido niveles de referencia con fines regulatorios aplicables a los materiales NORM. Pese a esto, algunos autores de los trabajos citados sobre mediciones en la República Argentina adoptan niveles de referencia para exención y limites de dosis que esta autoridad ha establecido para la industria nuclear en la NSBR[31]. En otros casos los autores utilizan como referencia límites de dosis equivocados y obsoletos[23-29].

- La NBSR de la ARN en su alcance e intervenciones menciona situaciones en las que estas normas serían aplicables más allá del ámbito de la actividad nuclear.

- Con respecto a las otras industrias de nuestro país que, al igual que el gas y el petróleo, también concentran tecnológicamente los radionucleidos naturales, sería conveniente realizar un estudio similar sobre las mismas.

- La ARN cuenta con todo el equipamiento y metodología de laboratorio necesarios para realizar estas mediciones y evaluación de escenarios reales en estas industrias.

- Finalmente, es fundamental aunar criterios, ya que se ve claramente que no se conoce la situación, y los escasos datos disponibles en nuestro país se tratan con criterios de referencia diversos y poco claros. A medida que este tema cobra importancia a nivel mundial, es necesario tener una visión clara del tema y adoptar una postura.

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PARTE II

RESÚMENES DE PUBLICACIONES EN REVISTAS

HIERRO Y ÓXIDO NÍTRICO EN PRECURSORES NEURONALES EXPUESTOS A LA IRRADIACIÓN γ*

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El objetivo del presente trabajo fue establecer el efecto de la irradiación γ en el contenido de Fe y óxido nítrico (NO) de células precursoras neuronales. Este modelo *in vitro* reproduce adecuadamente datos previos en fetos de ratas irradiados *in utero* obtenidos con modelos de irradiación *in vivo*. Cultivos primarios de células de cerebro de fetos de ratas Wistar en el día 17 de gestación se obtuvieron según procedimientos previamente descriptos (NeuroToxicology 2004; 25:387-398). Las células fueron irradiadas con una dosis de 2 Gy usando una fuente gamma de ^{60}Co (Gamma-Cell). El contenido total celular de Fe fue medido a través de la formación del complejo Fe^{2+}-batofenantrolina, previa mineralización. El *pool* celular de Fe lábil fue determinado por espectrometría de resonancia paramagnética a 77 K. El contenido de nitritos y nitratos en el medio de cultivo fue determinado por reacción de Griess y el contenido celular de NO fue evaluado empleando la sonda fluorescente 4,5 Diaminofluorescein diacetate. El contenido total de Fe en células controles fue de 2.91±0.06 nmol/mg prot y de 3.6±0.6 y 5.3±0.31 nmol/mg prot al cabo de 2 y 4 h pi. El *pool* de Fe lábil en las células controles fue de 163 ±33 pmol/mg prot y de 44±36 y 591±60 a las 2 h y 4 h pi, respectivamente. El contenido celular de NO alcanzó un valor máximo al cabo de 1 h pi (181% del control) para recuperar los valores iniciales al cabo de 2 h pi. El contenido de nitratos y nitritos en el medio de cultivo de células controles resultó de 0.07±0.01 μmol/mg prot y en el medio de células expuestas fue de 0.31 ± 0.03 y 0.43±0.05 μmol/mg prot a las 2 y 4 h pi, respectivamente. Se verificó un aumento significativo de la velocidad de generación de nitratos y nitritos en el medio al cabo de 1 h pi (0.03±0.01 y 50±4 μmol/min/mg prot para medios de células controles e irradiadas, respectivamente).

Estos resultados sugieren un aumento en el contenido de Fe al cabo de 4 h de la exposición de las células a la radiación γ probablemente debido a un aumento en la captación de Fe por una alteración en la funcionalidad de la membrana celular y este aumento se refleja en el *pool* de Fe lábil. Sin embargo al cabo de 2 h pi a pesar de mantenerse el contenido total de Fe de las células controles, se verifica una disminución del contenido de Fe lábil que podría deberse al aumento en el contenido de NO durante la primera hora pi, dado que el NO es un efectivo quelante de Fe.

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GENETIC AND EPIGENETIC FEATURES IN RADIATION SENSITIVITY
Part I: Cell signalling in radiation response*

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Recent progress especially in the field of gene identification and expression has attracted greater attention to genetic and epigenetic susceptibility to cancer, possibly enhanced by ionising radiation. It has been proposed that the occurrence and severity of the adverse reactions to radiation therapy are also influenced by such genetic susceptibility. This issue is especially important for radiation therapists since hypersensitive patients may suffer from adverse effects in normal tissues following standard radiation therapy, while normally sensitive patients could receive higher doses of radiation offering a better likelihood of cure for malignant tumours. This paper, the first of two parts, reviews the main mechanisms involved in cell response to ionising radiation. DNA repair machinery and cell signalling pathways are considered and their role in radiosensitivity is analysed. The implication of non-targeted and delayed effects in radiosensitivity is also discussed.

Keywords: Ionising radiation – Cell signalling – DNA repair – Radiosensitivity

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GENETIC AND EPIGENETIC FEATURES IN RADIATION SENSITIVITY
Part II: implications for clinical practice and radiation protection*

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Recent progress especially in the field of gene identification and expression has attracted greater attention to the genetic and epigenetic susceptibility to cancer, possibly enhanced by ionising radiation. This issue is especially important for radiation therapists since hyper-sensitive patients may suffer from adverse effects in normal tissues following standard radiation therapy, while normally sensitive patients could receive higher doses of radiation, offering a better likelihood of cure for malignant tumours. Although only a small percentage of individuals are "hypersensitive" to radiation effects, all medical specialists using ionising radiation should be aware of the aforementioned progress in medical knowledge. The present paper, the second of two parts, reviews human disorders known or strongly suspected to be associated with hypersensitivity to ionising radiation. The main tests capable of detecting such pathologies in advance are analysed, and ethical issues regarding genetic testing are considered. The implications for radiation protection of possible hypersensitivity to radiation in a part of the population are discussed, and some guidelines for nuclear medicine professionals are proposed.

Keywords: Radiation sensitivity – Radiation protection – Genetic disorders – Epigenetic disorders

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INVESTIGATION OF THE TL AND RL OF $KMgF_3$:La AND K_2YF_5:Pr^{3+} CRYSTALS IN ORDER TO ASSESS THEIR USE FOR IN-VIVO AND REAL TIME DOSIMETRY IN RADIOTHERAPY [*]

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The radioluminiescence (RL) and thermoluminescence (TL) properties of both $KMgF_3$:La and K_2YF_5:Pr^{3+} compounds have been investigated in order to evaluate their use for real time and in-vivo dosimetry in radiotherapy. The results are compared to those obtained employing LiF:Mg,Ti, The experimental data shows that the compound having the higher TL signal has the lower RL signal, and vice versa, i.e., $KMgF_3$:La bears the highest TL signal and the lowest RL signal, while K_2YF_5:Pr^{3+} bears the highest RL signal and the lowest TL signal. LiF:Mg,Ti shows RL and TL signals in between.

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