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PARTE I

Publicaciones y trabajos enviados a Congresos y/o Seminarios

Peak Efficiency Calibration for Compressed Disc Filter Sources by Using Point-Sources Methods

Aguiar, J.C. and Santini, E.S.

PEAK EFFICIENCY CALIBRATION FOR COMPRESSED DISC FILTER SOURCES BY USING POINT-SOURCES METHODS*

Aguiar, J.C. and Santini, E.S.

Nuclear Regulatory Authority
Argentina

An analytical formula for the full-energy peak efficiency for a 3M filter compressed disc sample, based on a modified expression for point sources on a range of energies from 59.5 to 835 keV, is derived.

- Radial measurements on a HPGe detector were made assuming rotational symmetry.
- The source activity concentration is considered to be homogeneous.
- A term for the photon self-absorption is included in the calculation.

* Disertación oral con presentación power point.

Assessment of the Main Deviations Found during Regulatory Inspections in Industrial Gamma Radiography in Argentina

Alonso Jiménez, M.T. and Ermacora, M.G.

Assessment of the Main Deviations Found during Regulatory Inspections in Industrial Gamma Radiography in Argentina

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Abstract. The present paper shows the regulatory work made through inspections concerning radiation safety and security in the area of industrial gamma radiography. An account is shown of the items which should be available in the storage facility where the equipment and radioactive sources are placed as well as in site radiography and which are required to the users of the equipment meant for this practice, in order to ensure compliance with the regulations in force. The objective of this paper is to assess the most frequent deviations to the regulatory standards found during this type of inspections, such as lack of complete and operable radiation protection equipment, lack of available elements bound to act in case of emergency, no delimitation of controlled areas, gamma radiography equipment stored in unlocked position, wear gauge unavailable, record of movements of equipment and radioactive sources not updated, anomalous monthly or annual personal dose values, among the main findings. In particular, in connection with dose reports, some cases of occupational exposed workers exceeding the annual dose limits or investigation levels are highlighted. In addition, the explanations provided by the person responsible for radiation protection of companies in regard to the possible causes or circumstances which might have given rise to such dose values, the conclusions reached and the preventive and corrective measures implemented by these companies are analysed. Finally, an assessment of the annual dose distributions of the occupational exposed workers concerned with this kind of practice in the period 2004-2007 is shown. As a conclusion, the relevance of these deviations according to degradation in safety and security performance and the role of the regulatory work as a promoter of the implementation of safety culture will be emphasized.

KEYWORDS: *regulatory inspections, industrial gamma radiography, radiation safety, occupational exposed workers, dose distributions*

1. Introduction

Among non-destructive tests, industrial gamma radiography is the technique which is controlled by the Nuclear Regulatory Authority (ARN) because of being the one which utilizes radioactive material. Among the radioisotopes used for this technique in the world -¹⁹²Ir, ⁶⁰Co, ¹³⁷Cs and ⁷⁵Se, among others-, the first mentioned is the most commonly used in Argentina.

The objective of this paper is to show the regulatory work made through inspections concerning radiation safety and security in the area of industrial gamma radiography, assess the most frequent deviations to the regulatory standards found during this type of inspections, highlight some cases of occupational exposed workers exceeding the annual dose limits or investigation levels, analyse the explanations provided by the person responsible for radiation protection (PRRP) of companies in regard to the possible causes or circumstances which might have given rise to such dose values, the conclusions reached and the preventive and corrective measures implemented by these companies, assess the annual dose distributions of the occupational exposed workers concerned with this kind of practice and emphasize the relevance of the deviations mentioned according to degradation in safety and security performance and the role of the regulatory work as a promoter of the implementation of safety culture.

2. Regulatory Framework

The ARN was established as an autonomous body reporting to the President of Argentina by Act 24,804 known as the Nuclear Activity National Act, which came into force on April 25, 1997, and is empowered to regulate and control the nuclear activity with regard to radiation and nuclear safety,

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physical protection and nuclear non-proliferation issues [1]. Decree 1390/98 regulating said Act defines its scope and procedures facilitating its enforcement.

In its capacity as the national authority, the ARN grants licenses, authorizations and permits, as appropriate. In addition, the ARN performs control activities to ensure that persons responsible for each practice comply with the provisions set forth in the standards and other regulatory documents [2].

In the particular case of industrial gamma radiography, the requirements which must be fulfilled concerning radiation safety are specified in the Standard AR 7.9.1. "Operation of industrial gammagraphy equipment" [3]. Other related standards in force are: AR 7.11.1. "Individual permits for industrial gammagraphy equipment operators" [4], AR 10.16.1. "Transport of radioactive materials" [5], AR 10.1.1. "Basic Radiation Safety Standard" [6] and AR 10.13.2 "Standard on the physical protection of sealed sources" [7].

In case of non-compliance with the regulations in force, the Sanction Regime is applied. [8]

3. Regulatory inspections

An account is shown of the items which should be available in the storage facility where the equipment and radioactive sources are placed as well as in site radiography and which are required to the users of the equipment meant for this practice, in order to ensure compliance with the regulations in force.

3.1 Storage facility inspections

During the inspections carried out to the storage facilities of the industrial gamma radiography companies, the following items are verified:

3.1.1 The record book and documentation, which should include:

- Inventory of projectors, containers and radioactive sources, which should correspond with the Operating License specifications. It is contrasted with the database of the ARN, which is updated with the notifications of equipment and source transfers that the operating organization must submit;
- Movement of projectors and sources, in operation;
- Radiation protection equipment in accordance with the quantity of gamma radiography equipment in inventory, as well as its calibration;
- Maintenance record of the gamma radiography equipment;
- Personal dosimetry monthly record, with the corresponding notification signed by the involved personnel;
- Sources, projectors and type B(U) package certificates;
- Operational, emergency and transport procedures.

3.1.2 The store of the equipment

- Correct warning signs;
- Appropriate physical protection and radiation safety.

3.1.3 The equipment

- Dose rates close to the projectors surface;
- Correct identification and labeling of the projector container as well as the source housed in it;
- Source assembly connector identification, verifying its coincidence with the source certificate issued by the manufacturer;
- Lock mechanism;
- Source assembly connection to drive cable, using a wear gauge;

- Guide tubes and other accessories.

3.1.4 The radiation protection equipment

- Quantity as specified in the regulations;
- Good working condition;
- Calibration.

3.2 Site radiography inspections

During the site radiography inspections, some of the above-mentioned controls are carried out, and in addition, the following items are verified:

3.2.1 The workplace

- Warning notices and barriers of the area;
- Dose rates at the boundary;
- Collimators, local shielding and emergency kit.

3.2.2 The workers

- Compliance with the operational procedures, placing particular emphasis on the verification of the correct source retrieval into the shielded container by the operator;
- Intervention of a minimum of two people in the operation (one operator and one assistant), verifying that at least one of them possesses an individual permit issued by the ARN;
- Correct use of the radiation protection equipment by the operator and his assistant.

3.2.3 The transport

- Correct labeling according to the measured dose rates;
- Compulsory documentation;
- Warning notices in the vehicles and appropriate fasteners.

4. Most frequent deviations found during regulatory inspections

4.1 Non-conformities found during inspections carried out to storage facilities in the year 2007

4.1.1 The records and documentation

- Records not updated;
- Lack of notification to the ARN in relation to changes of storage facilities and legal addresses, increase in inventory, etc., and the corresponding submission of the License modification form;
- Lack of type B(U) package certificate.

4.1.2 The store of the equipment

- Unsuitable anchoring of the metallic box to the site floor;
- Modifications to the conditions of radiation and physical safety;
- Improper warning notices.

4.1.3 The equipment

- Improper maintenance of the gamma radiography projector;
- Unlocked projector containers;

- Illegible labels.

4.1.4 The accessories

- Lack of wear gauges;
- Wear gauge test of drive cable not passed;
- Lack of collimator;
- Incorrect identification of control cables and guide tubes.

4.1.5 The sources

- Radioactive sources not compatible with projector;
- Possession of decayed radioactive sources;
- Incorrect source identification;
- Lack of notification of source changes.

4.1.6 The radiation protection equipment

- Lack of the complete radiation protection equipment according to the quantity of gamma radiography equipment as specified in the Operating License.

4.1.7 The dosimetry

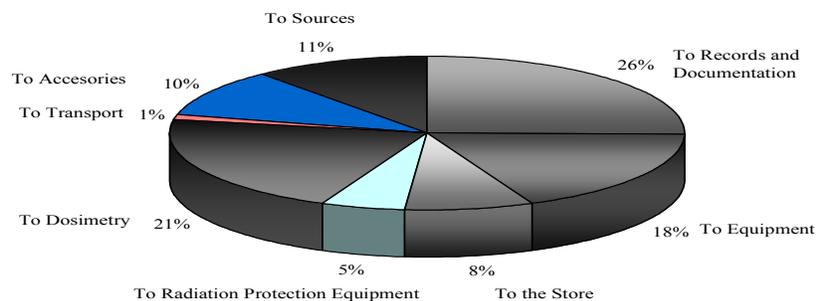
- Lack of: submission of the annual dose report, updated dosimetry, updated access by workers to their own dose records and the required dosimetry service contracts;
- Higher than the expected monthly dose values for this practice;
- Lack of personal dosimeters identification.

4.1.8 The transport

- Incorrect warning notices in the transport vehicle.

In Fig. 1, a detail of the aforementioned findings, in percentage of occurrence, can be appreciated.

Figure 1: Non- conformities found during inspections in the storage facilities (period: 2007)



4.2 Non-conformities found during site inspections carried out in the period 2004- mid-2008

4.2.1 The equipment

- Projector containers in unsuitable conditions (e.g., broken lock mechanism, illegible labels);
- Wear gauge test not passed.

4.2.2 The workplace

- Lack of complete set of elements for source recovery.

4.2.3 The workers

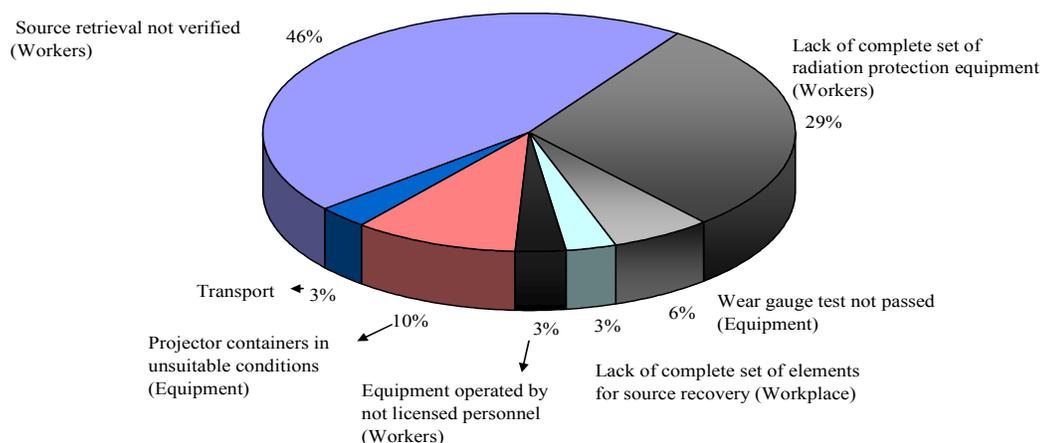
- Source retrieval not verified;
- Equipment operated by not licensed personnel (e.g., assistant);
- Lack of complete set of radiation protection equipment.

4.2.4 The transport

- Non-compliance with transport requirements (unlocked equipment inside the vehicle, equipment not suitably fixed to the vehicle).

In Fig. 2, a detail of the above-mentioned findings, in percentage of occurrence, can be appreciated.

Figure 2: Non-conformities found during site inspections (period: 2004- mid-2008)



In order to address the non-conformities, the ARN establishes the corresponding requirements through the Inspection Report, setting a deadline for their compliance, which is controlled by the assigned inspectors. In case of non-compliance with the imposed deadlines, the corresponding procedures leading to the application of sanctions are applied, according to the regulations in force.

5. Assessment of dose reports

In the Figs. 3, 4, 5 and 6, a graph including the annual dose distribution of the occupational exposed workers involved in industrial gamma radiography and estimates of the maximum and average annual dose in the last years is shown.

Figure 3: Distribution of doses reported corresponding to the year 2004

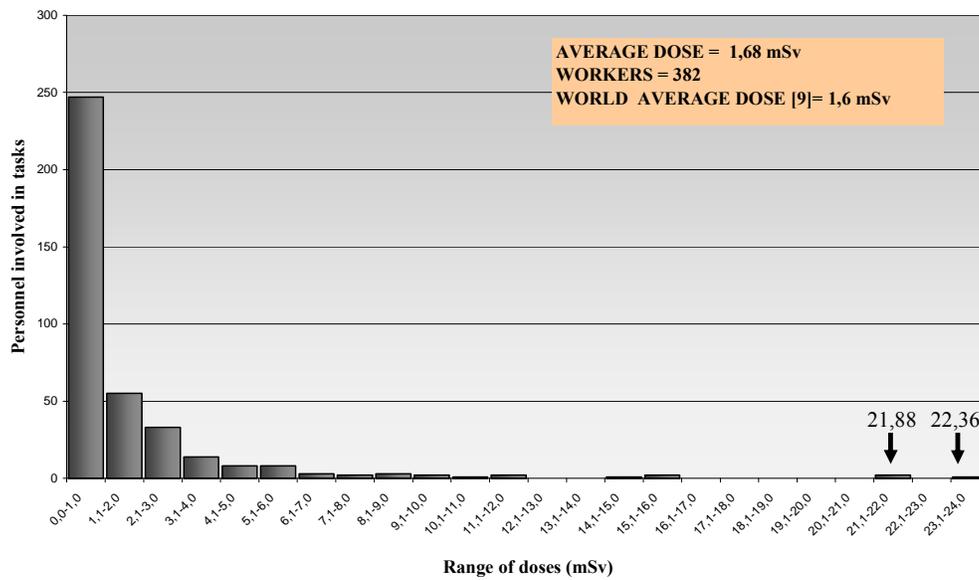


Figure 4: Distribution of doses reported corresponding to the year 2005

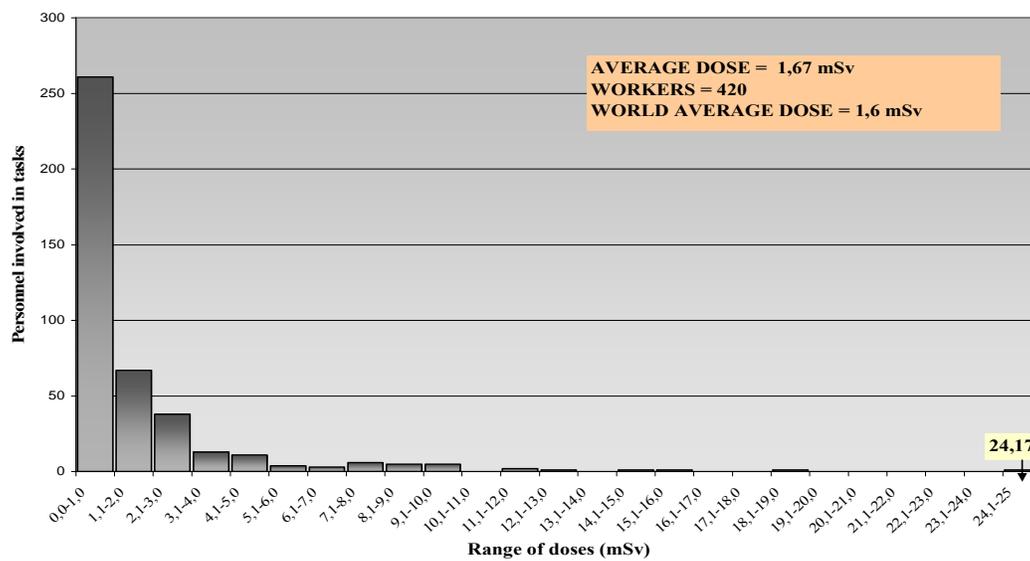


Figure 5: Distribution of doses reported corresponding to the year 2006

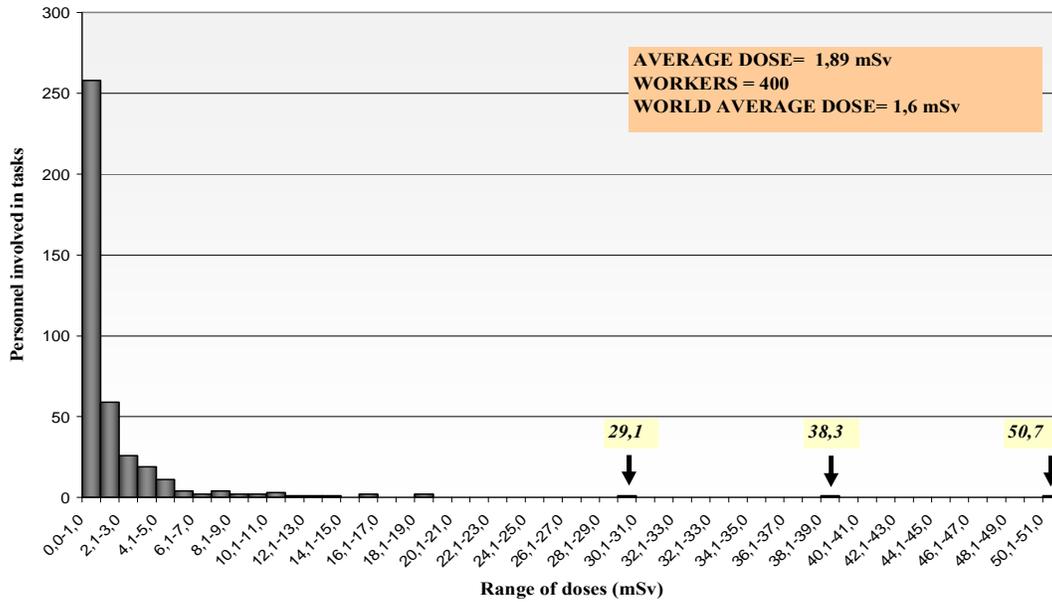
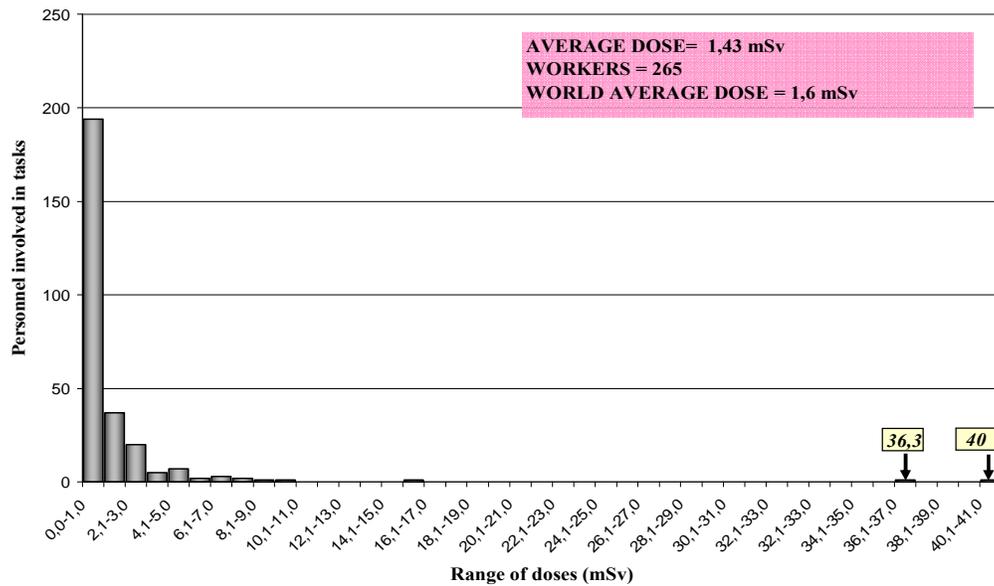


Figure 6: Distribution of doses reported corresponding to the year 2007



This last graph has been drawn with the data available at the time of the preparation of the present paper, thus not containing complete information.

The cases of four companies in which some of their occupational exposed workers dose reports exceeded the annual dose limits or investigation levels are analyzed below.

5.1 Case A

Eleven workers had records of higher than the expected values of doses for this practice.

The ARN conducted an investigation of this company and requested the PRRP to make a report analyzing the operation situations, which was elaborated with the help of the personnel having presumably received higher than the expected values of doses. This report also had to include details of the work places, periods and working teams (radiographer and assistant).

Eight workers out of eleven declared that they did not know the cause which might have given rise to such dose values, two of them did not declare at all and only one of them stated that he might probably have left his personal dosimeter out of place. The PRRP stated that the possible cause might have been a shipment of the dosimeters in close contact with the source container when the sources were transported from the importer to the work area of the company.

From the investigation nothing could be concluded in relation to the occurrence of possible incidents, due to the fact that those values of doses had taken place in different scenarios (companies, provinces and working teams), and therefore certain causes for those doses could not be found. One possible explanation for the 7,7 mSv, 9 mSv and 10 mSv might be a great amount of operations, with prolonged exposures, excess of work and negligence in taking suitable distances and/or use of shielding, i.e., non-compliance with the operational procedures. For the cases of monthly dose values of 17 mSv, 29 mSv, 38 mSv and 50 mSv, they could be attributable to an incident, or to an overexposure of the personal dosimeters for having carelessly left them close to the source, as the PRRP stated.

The ARN requested the PRRP to resign to his responsibility in the company. In addition, the operating organization was requested to submit to the ARN weekly notifications containing detailed information in relation to the places where the radiographic exposures would be carried out, one week in advance of their taking place, in order to be able to intensify the control by means of site inspections.

5.2 Case B

One worker had a record of a monthly dose of 36, 3 mSv.

The explanation provided by the radiographer was that he had left his personal dosimeter inside a case, which inadvertently remained in the radiation area.

The radiographer communicated this event to the PRRP and logged it in the record book of personal dosimetry. In addition, the assistant dose was logged, who did not receive any overexposure.

The PRRP notified the ARN about the implementation of a Quality Management System. As a result, this non-conformity and the corresponding corrective action would be logged, and the PRRP would organize a meeting with all the personnel during which the precautions to take in order to prevent the occurrence of similar events in the future would be emphasized.

5.3 Case C

One worker had a record of a monthly dose of 40 mSv.

The explanation provided by the radiographer was that his personal dosimeter fell down in the area of work, close to the radiation source, and he noticed it after the second exposure.

As a conclusion, the PRRP committed himself to constantly control the presence of the dosimeter, and in particular, paying careful attention previous to each radiographic exposure.

5.4 Case D

Three workers had a record of higher than the expected values of doses for this practice.

The explanation provided by one of the radiographers was that he had lost his personal dosimeter for four or five days, that is why he inferred that it could have been left close to the radioactive source.

The PRRP committed himself to reinforcing the training of his personnel in order to correct possible mistakes in the operations. He also assumed the responsibility for assessing carefully each one of the radiographers and assistants, for exchanging the working teams, carrying out unannounced controls in the places of work in order to evaluate the radiographic exposures and verifying the performance of his personnel for correct radiation protection measures.

From all of the above-mentioned issues, some considerations can be extracted:

- Negligence in the use of personal dosimeters by the personnel involved in the tasks;
- Lack of knowledge concerning the use of personal dosimeters by the personnel;
- Lack of commitment to radiation protection matters by the PRRP;
- Lack of implementation of a Quality Assurance Programme;
- Lack of communication of incidental or bad practice situations to the ARN.

6. Conclusion

Radiation safety in industrial gamma radiography depends on human performance as well as on the intrinsic safety of the radiographic equipment.

Inspections reflect the safety degree of the operating organization at the moment they are carried out; however, this may not show the actual situation. A fluid communication between the Regulatory Authority and the companies is essential.

The deviations observed during inspections are related to non-compliance with regulations and working procedures, which is indicative of degradation in safety and security performance.

Operating organizations should provide workers with adequate and complete radiation protection and safety equipment and conduct a periodic review of the overall effectiveness of the protection and safety measures.

It is crucial that the personnel involved in the tasks comply with a continuous training programme in the operating organization, in addition to the courses attended to obtain and renew the Individual Permits. Radiographers as well as their assistants should be aware at all moments of the consequences to health that any non-compliance with the working rules may have.

In relation to dose assessments, the annual dose distributions are the expected for this kind of practice. The higher than normal values observed may be attributed to improper use of dosimeters and/or non-compliance with operational procedures, according to the investigations carried out.

It is important to highlight the relevance of the Regulatory Authority role in conducting investigations in relation to the causes of dose reports exceeding the expected values for this type of practice, as well in communicating the lessons learned from the evaluation of broken safety barriers and from anomalous situations, in order to contribute to the promotion of safety culture.

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Incident in Industrial Gamma Radiography in Argentina

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Incident in Industrial Gamma Radiography in Argentina

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Abstract. In the present work a description of an industrial gamma radiography incident is made and an analysis of the events which led to it is presented. The event took place during a routine regulatory inspection carried out by personnel of the Nuclear Regulatory Authority in the year 2004 to a storage facility of projectors and containers belonging to a company licensed to operate gamma radiography equipment. On arriving to the store authorized by the ARN, placed in an industrial site, it was verified that it was empty and without surveillance. The inspectors contacted the person responsible for radiological protection, who went to the place and declared that the totality of the equipment and radioactive sources had been moved to a private country house. The reason for this change, according to the statement, was that the company had gone bankrupt and the authorized storage facility had remained without surveillance and without personnel in charge. In the country house, inside the room meant as a store, high environmental equivalent dose rates were detected. After verifying the dose rates in each of the projectors, it was checked that one of them measured high dose rates. For this reason, it was isolated and the source recovery was done, due to the suspicion that it had remained in a non-shielded position. After projecting it several times towards a lead container due to the fact that high dose rates were still being measured, the person responsible for radiological protection proceeded to extract the source from the guide tube, finding that it had been transferred in the reverse order. According to the record of movements of equipment and radioactive sources of the company, the source transference had been done, in the previous store in the month of March 2004, by a radiographer who did not work at the company any longer and whose permit had expired. The results of the estimations of the doses received during the process of the source recovery are presented. Finally, an assessment relating to the circumstances that led to the event, its consequences and an analysis of the non-compliance with the Argentine regulations concerning radiological protection is made.

KEYWORDS: *incident, gamma radiography, radioactive source*

1. Introduction

In Argentina there is an average of 60 companies in the last years, which provide industrial gamma radiography service, one of the techniques used as non-destructive tests. In the area of industrial gamma radiography, the rhythm of work is associated with construction works, enlargement of pipelines and other works of the industrial sector. For this reason, there are periods of low activity and at the same time less production and fewer imports of radioactive sources; there are companies that interrupt the operation of their gamma radiography equipment, reduce the source exchanges and terminate contracts with radiographers and assistants. In these periods, as a consequence of economic and administrative difficulties, negligence tends to be committed which can attempt against the radiological safety of the involved occupational exposed workers as well as members of the public. It is important to emphasize that it has not occurred any other accident related to the technique of gamma radiography which have resulted in high doses to occupational exposed workers or members of the public in our country since the accident that took place in the year 1968 [1].

The Nuclear Regulatory Authority (ARN), as an autonomous body reporting to the President of Argentina by Act 24,804 known as the Nuclear Activity National Act, which came into force on April 25, 1997, is empowered to regulate and control the nuclear activity with regard to radiation and nuclear safety, physical protection and nuclear non-proliferation issues [2]. Among its functions, it imposes sanctions which are graduated according to the gravity of the infraction in: written warning, fine which shall be applied proportionally to the gravity of the infringement and according to the damage potential, suspension of a license, permit or authorization or its revocation. In view of serious infringements of the radiation safety standards, it can confiscate the nuclear or radioactive materials, as well as precautionary close the installations subject to regulation. As the Nuclear Activity National

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Act clarifies, it is understood by serious infringement the non-compliance which implies a serious threat to the security of the population or the protection of the environment or when the application of the physical protection or safeguards measures can not be guaranteed.

The area of industrial gamma radiography of the ARN belongs to the Control of Radioactive Type II and III Facilities Division, and this one to the Radiological and Physical Protection and Safeguards Department. One of the main functions of the aforementioned area is to conduct inspections to all the authorized storage facilities of the gamma radiography companies with an annual frequency, and to other temporary storages of the operating organizations, as well as site radiography inspections which can be coordinated with Hygiene and Safety personnel of the clients of non-destructive tests, without previous announcement to the gamma radiography companies. The present paper gives a description of an event which took place during one of the routine inspections to an authorized store of an operating organization, where an anomaly was detected which resulted in the exposure of some people. The cause of this anomaly was the incorrect transference of a decayed source of ^{192}Ir . The correct transference of such source is described and a dosimetric assessment is made. A detailed analysis considering other scenarios can not be made due to the high degree of uncertainty that this initiating event entails. An assessment relating to the circumstances that led to the event, its consequences and an analysis of the non-compliance with the Argentine standard concerning radiological protection is made.

2. Some data of the company

The operating organization to which this work refers had its legal address and its storage facility address in an inland province of the country. At the moment of the incident the aforementioned company had more than 20 gamma radiography projectors and containers. The authorized store was placed in an industrial site which was under 24-hour surveillance. It is worth clarifying that this organization, according to the previous year inspection, complied with all the requirements specified in the national regulations in force. (AR Standard 7.9.1.) [3].

3. Description of the event

The inspection team (consisting of two industrial gamma radiography inspectors) turned up in the industrial site on September 30, 2004, in order to conduct the annual routine inspection, finding the organization store closed with padlock and without surveillance staff. From that place, the inspection team contacted the Person Responsible for Radiological Protection (PRRP) of the organization, who immediately went to the storage facility. Once there, he declared that the totality of the equipment and radioactive sources had been moved to a private country house, place which had formerly been an authorized store by the ARN.

In the country house, all the equipment was placed in a metallic box closed with a padlock, inside a store made of concrete, covered with a roof, with a door closed by padlock. It called the attention of the inspectors the environmental equivalent dose rate measured from the door of the store ($50 \mu\text{Sv/h}$), two metres away from the box.

The two inspectors started the verification of each one of the projectors separately, outside the store. The last projector to be controlled was brand Tech Ops, model 533, with no visible serial number, which was being used as a container of a decayed source of ^{192}Ir , with an activity of 25,4 GBq. The radiation monitor of one of the inspectors emitted a warning audible signal and a dose rate of approximately 15 mSv/h was measured at 0,5 m. The operator was requested to move the projector away and to use the available shielding in order to cover it until the situation had been assessed. Not being source recovery equipment (shielding and long handling tongs) available in the country house, the operator proceeded to move the projector away and shield with things available in site (metal sheets, wood boxes, among others).

An imbalance in the uranium shielding was suspected and it was requested to another operating organization located in the proximity to lend an empty lead container in order to house that source.

The source transference from the projector to the lead container was started using a control cable for Tech Ops projectors which the PRRP had at hand in the store. The pigtail identification was verified, distinguishing the inscription “engage gauge”, which corresponds to a dummy. However, the high dose rates measured made it unthinkable that it could be a dummy. After three attempts of source transference, the same high dose rates were still being measured. Subsequently, the guide tube was disconnected from the projector, it was moved away and the dose rate measured at the projector dropped to 20 $\mu\text{Sv/h}$, which was consistent with the uranium shielding of the projector. The high dose rates continued to be measured in the connection of the guide tube and the container, which did not seem reasonable if the source was inside it. The PRRP then suggested that the source could have been put in reverse order, not correctly transferred. Considering this possibility, he disconnected the guide tube from the container and after verifying that the suspicion was confirmed, the PRRP took the source out of the guide tube with a tong and put it inside the shielding. The dose rate measured at this time was 20 $\mu\text{Sv/h}$ in the container sides and 1 mSv/h at the top of it. The PRRP plugged up the container and put the projector away in an overpack of the company.

By way of illustrating, a cross-section view of a Tech Ops projector with the position of a source abutting the dummy connector is presented in Fig.1 and a cross-section view of a Tech Ops projector with the source in a fully-shielded position is shown in Fig. 2.

Figure 1: Cross-section view of the projector with the source as it was detected during the inspection.

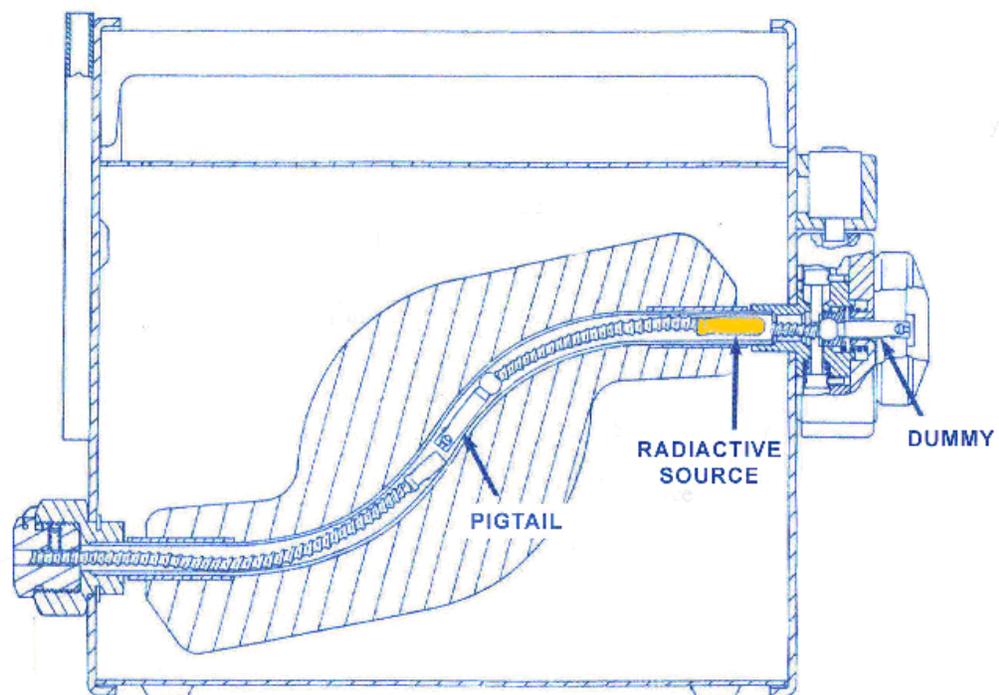
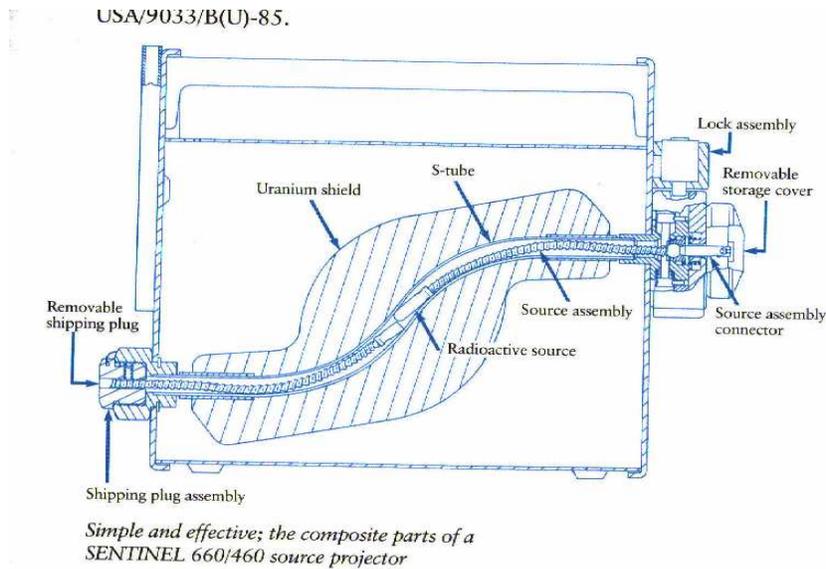


Figure 2: Cross-section view of the Tech Ops projector with the source in a fully-shielded position.



4. Subsequent actions

Having then verified that the projector shielding was not damaged, the correct source transference to the projector was made by two authorized radiographers belonging to another operating organization and under the supervision of the audited one.

The ARN area manager was notified of the event and he requested that the country house was under surveillance by the Federal Police. By common consent with the PRRP of the operating organization, all the equipment and sources were transferred to the Waste Disposal Area of the Atomic Energy National Commission, and both the Operating License of the company and the Individual Permit of the PRRP were suspended.

As well as the suspensions, the ARN applied both to the operating organization and the PRRP the Sanction Regime [4] in force. As part of it, the PRRP was requested to send a report containing a description of the facts which led to this situation, part of which is mentioned below.

In his statements, the PRRP informed that at the beginning of March 2004, having decided the sale of one empty projector, he requested one person of the staff (whose individual permit had expired in the year 2001) to empty this projector by transferring the source contained in it to the above-mentioned projector Tech Ops 533.

The PRRP declared that once the transference had been finished, and recognizing that it had not been verified by him, the projector was stored in a pit at a depth of 0,60 m, with concrete walls, in a place located furthest from the access door. He stated that the said projector was not visible from the bunker door and neither was it visible from the place where the projectors of frequent use were kept. He also assumed all the responsibility for not having verified the correct source transference to the projector used as container.

In addition, the PRRP justified the sudden move stating that it was due to an unfavourable situation of the company. For this reason, the company stopped providing gamma radiography services in the middle of July and stopped offering services as a company in the month of September.

The PRRP stated that when he began cancelling the contracts of the company staff, the industrial site remained without surveillance, and because it was near a marginal zone, he thought it would be convenient to put the equipment under shelter in the previous company store (country house), which was under surveillance by a member of his family during the daytime.

The PRRP said he had cancelled the dosimetric service from March 2004; for this reason, there were not records of the company staff monthly doses.

After the recovery task, the company records were verified, finding the transference from one projector to the other one and the name of the person who had made it, registered by the PRRP and not signed by the person who effectively made it. Once located this former radiographer, during his defence, he denied emphatically having made such source transference.

The inspectors dosimeters were immediately evaluated and the one from the inspector "1" gave a result of 0,1 mSv and the one from the inspector "2" measured less than the detection limit (0,1 mSv).

The ARN imposed fines both to the operating organization and the PRRP, for what had happened. In relation to the person who had presumably made the transference, considering his defence, in which he denied all responsibility concerning such source transference, and having not found enough proofs, he remained exempt from sanctions.

5. Dosimetric assessment of the source recovery

At the moment of the source recovery there was no radiation protection equipment or suitable source recovery equipment at the site. There were not personal dosimeters due to the fact that the operating organization had cancelled the contract with the provider of this service once it gave up operating.

5.1 Data

Place: Private country house. Date of ARN inspection: September 30.

Source activity: 4,181 GBq; Calibration date: 25/02/03

Activity at the moment of the inspection: 25, 4 GBq. In order to simplify and make the calculations more conservative, a naked source is considered.

The distances are estimates and the length of the projector, the control cable and the guide tube are considered.

Persons involved:

- PRRP of the organization
- Member of the public (PRRP's relative)
- Inspector "1"
- Inspector "2"

5.2 Calculations

$$D = \frac{A \cdot \Gamma \cdot t}{d^2}$$

D: dose

A: activity

Γ : Gamma factor is the absorbed dose rate in mSv.h⁻¹ at 1m from 1 GBq of the radionuclide. In case of ¹⁹²Ir is 0, 13.

t: time

d: distance

Table 1: Estimation of doses for the PRRP considered by tasks.

Task	Description	Duration	Distance source-operator	Dose (mSv)
1	The projector is carried to a table for inspection	10 s	15 cm (hand and leg)	0,408
2	The projector is transferred to a remote place	45 s	15 cm (hand and leg)	1,834
3	The projector is carried to the recovery zone	45 s	15 cm (hand and leg)	1,834
4	The control cable is connected to the pigtail	40 s	7 cm (hand)	7,488
5	The guide tube is connected	15 s	29 cm (hand)	0,164
6 (a)	Process of transference	1 min	9,29 m (whole body)	0,001
7	Verification of the connection control cable- pigtail	1 min	2,29 m (hand)	0,010
8	Control of source position	2 min	2 m (whole body)	0,028
9	Disconnection of the guide tube from the projector	15 s	2 m (hand)	0,003
10	Recovery	15 s	15 cm (hand)	0,610

^(a) In this item, the various projections that were made are considered (in total: three projections)

Table 2: Estimation of doses for the inspector “1” in the whole body, due to the tasks carried out by the PRRP which kept her more exposed to the source.

Task	Description	Duration	Distance source-inspector	Dose (mSv)
1	The projector is carried to a table for inspection	10 s	30 cm	0,102
3	The projector is carried to the recovery zone	45 s	1 m	0,041
4	The control cable is connected to the pigtail	40 s	1 m	0,037
5	The guide tube is connected	15 s	1,29 m	0,008
6 (a)	Process of transference	1 min	9,29 m	0,001
7	Verification of the connection control cable-pigtail	1 min	2,29 m	0,010
8	Control of source position	2 min	2 m	0,028
9	Disconnection of the guide tube from the projector	15 s	2 m	0,003
10	Recovery	15 s	2 m	0,003

^(a) In this item, the various projections that were made are considered (in total: three projections)

Table 3: Estimation of doses for the inspector “2” in the whole body, due to the tasks carried out by the PRRP.

Task	Description	Duration	Distance source-inspector	Dose (mSv)
1	The projector is carried to a table for inspection	10 s	30 cm	0,102
	All the operation of transference	434 s	10 m	0,004

5.3 Results

Considering all the tasks carried out during the inspection in relation to the source recovery, the estimated total dose for the PRRP of the organization was 12,35 mSv in limbs and 29 μ Sv in whole body.

The equivalent dose of "1", who was nearer the operator, was 233 μ Sv in whole body.

The equivalent dose of inspector "2" was 106 μ Sv in whole body.

The equivalent dose of the PRRP's relative (member of the public), who was close to the inspector "2", was 4 μ Sv.

6. Analysis of the situation

The operating organization committed a series of irregularities, detailed in chronological order:

- The PRRP ordered a person whose individual permit as operator had expired to make a source transference (in case the statements made by the PRRP are true).
- The operator in charge of the transference and/or the PRRP did not use a radiation monitor at the moment the source was transferred to the projector model 533, used as a container, in order to verify the correct source transference.
- The PRRP did not measure the radiation background in the store during all the period (March-September) that the source remained not fully shielded.
- Suspension of the dosimetry service, even being the operating organization with its Operating License in force and without notifying it to the ARN.
- The change of store was not notified to the ARN, prior to the transference of all the equipment with their contained sources.
- Lack of radiation protection equipment and source recovery equipment in the country house.

The situation could not be completely clarified. A record of the persons who operated the equipment and who made source transfereces between equipment was followed by the PRRP. The person suspected was logged in that record. However, there is nothing that can prove this person had effectively made the transference of this source. It is possible that the person who made the source transference from projector to projector did not know how to transfer a source and that it could be pushed by the security cap which is put to the projector.

The national standard [3] specifies in its item 31 that after each radiographic exposure (in this case it was not an operation, but a transference which generates radiation exposure) the operator must verify the correct source recovery to the projector (or container) through monitoring using a portable survey meter. The non-compliance with the said item is considered in the article 16° of the Sanction Regime [4]. This situation is further aggravated by the fact that this person (in case he had done it) had his permit expired [5].

It is considered that this non-compliance with the regulations in force could have resulted in considerable doses to occupational exposed workers and/or members of the public. For this reason, in connection to the severity of the infringement, it is considered **serious**, even when the good willingness shown by the PRRP to lessen the effects is considered an attenuating fact. In relation to the damage potential, it is also considered **serious**, due to the source category involved (industrial gamma radiography) and its potential to cause serious damages in people health.

In addition, the item 32 specifies that while gamma radiography projectors are not in use, they must be put away in the storage facility authorized by the ARN. The non-compliance with the said item is considered in the article 8° of the Sanction Regime.

In connection with the transference of all the equipment with their respective sources to the store which was not included in the Operation License as being authorized by the ARN, it is understood that the non-compliance is of an administrative type due to the fact that it did not imply a worsening of radiological safety. It is also considered that it was the first authorized store the operating organization had had and that the transfer was due to an assessment of the safety and security conditions, which had become not suitable for the authorized store at that moment.

7. Conclusion

It is proved, like in the majority of the cases of accidents and incidents related to this technique, that these situations are caused by the non-compliance with the regulations in force; for example in this specific case, the lack of use of radiation protection equipment and the excess of confidence of the PRRP in delegating and not controlling that the operation had been made in the correct way (considering that it had happened as he had declared).

Only the strict observance of the regulations in force, the continuous training and practice of the staff/personnel, the compliance with the procedures, and the strict control regarding all that is concerned with radiological and physical protection in the operating organization as far as the PRRP is concerned make it possible that situations like this one which attempt against individuals health do not occur.

It is important to highlight that all the irregularities which conducted to the above-mentioned unjustified exposures are indicative of a lack of safety culture within the operating organization.

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Facing the Facts: The Suspension of a Design Requirement Applicable to Nuclear Power Reactor Effluents

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Facing the facts: The suspension of a design requirement applicable to nuclear power reactor effluents

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Abstract. A design requirement aimed at limiting the discharge of globally dispersed long-lived radionuclides released by nuclear power reactors was in force in Argentina since 1979 till recently. The practical result of such regulatory measure was the need to retain C-14 in the PHWR under construction, as well as in future heavy water reactors to be built in the country. This paper explains the basis of such requirement, which was formulated as a collective dose constraint per unit of energy generated, and the main assumptions that triggered it. The differences between the assumptions made at that time and the reality of nuclear power generation at the beginning of the 21st century, as well as their implications in relation to the requirement are described, including the Suess effect and its impact in the total dose due to C-14. Finally, the facts that made no longer reasonable to keep in force the above-mentioned requirement are presented.

KEYWORDS: *C-14, Atucha II, nuclear power reactors, heavy water reactors, Suess effect.*

1. Introduction

By the end of 1979 a design requirement aimed at limiting the discharge of globally dispersed long-lived radionuclides produced by nuclear power reactors was in force in Argentina [1]. Such requirement was formulated as a limitation of the collective dose commitment per unit of electrical energy generated by a nuclear power reactor and the practical result was the need to retain C-14 in the Atucha II power reactor, as well as in future heavy water reactors to be built in the country.¹

In 2005, the Nuclear Regulatory Authority (ARN is the Spanish acronym and will be used in what follows) decided to re-asses the basis of such requirement and an ad-hoc task group was created to that end. The report of the task group was presented to the Board of Directors by the middle of 2007 and, as result of such report, the ARN decided to suspend the application of the above mentioned requirement (i.e. it is not more mandatory, even for the nuclear power reactor under construction, Atucha II).

This paper summarizes the main aspects of the task group report, addressing, in particular:

- The original basis of the collective dose constraint per unit of energy generated aimed at limiting the discharge of radioactive effluents of nuclear power reactors, including the scenarios at the time the requirement was established and the main assumptions that triggered it.
- The differences between the assumptions made by the end of 1970 and the reality of nuclear power generation at the beginning of the 21st century as well as their implications in relation to the requirement, including the limited contribution of nuclear power energy to the world demand of electricity during the last decades and the implication of the Suess effect on the per caput world dose due to C-14.
- The limitations of the international radiation protection system regarding globally dispersed radionuclides and the conclusions that lead the ARN to the suspension of this requirement.

2. Design requirements and C-14 retention

Regarding the design of the radioactive effluent control systems of nuclear power reactors, the ARN required that such systems shall be optimized and two dose constraints were established. These dose constraints were formulated as follows (translation of point 6 of the ARN standard AR 3.1.2. [2]):

“6. Retention should be such that the following dose constraints are fulfilled:

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¹ Heavy water reactors produce more C-14 per unit of energy generated than light water reactors.

- a) *The annual effective dose in the critical group due to the release of radioactive effluents shall not exceed 0.3 mSv.*
- b) *The collective effective dose² shall not exceed 15 man mSv per MWe³ of electrical energy generated.”*

Both dose constraints are optimization boundaries and the first one refers to the dose that would incur the critical group of a nuclear power reactor as result of its normal operation. The second one, as will be explained below, by restricting the accumulation in the biosphere of globally dispersed long-lived radionuclides, intends to limit the future per caput dose rate of the world population, and then, if universally applied, the global contribution to the annual dose rate of all critical groups [3]. In other words, the collective dose constraint per unit of energy generated was not aimed at limiting the dose of the critical group of a given installation, but of all critical groups in the distant future.

With this framework, the research carried out confirmed that regarding nuclear power reactors the only radionuclide of interest is C-14 since:

- It has a not negligible production rate by unit of nuclear energy generated, particularly in the case of reactors moderated and cooled by heavy water.³
- It is globally distributed.
- Its inventory in the biosphere increases continuously due to its extremely long half-life (5.700 years).

Consequently, a collective dose constraint of by unit of energy generated may imply the need to retain C-14 in a nuclear power reactor and, later on, to manage it as to assure its isolation from the biosphere during an appropriate period of time. Due to its long half-life, its disposal would have to be carried out in deep geological repositories, as it is foreseen for high-level waste.

One of the reasons that triggered the re-assessment was the observation that in spite of the fact that no nuclear power reactor has systems specifically intended to retain C-14, including those under construction,⁴ the concern for the generation of C-14 in nuclear power reactors persists [4], [5], [7], [8], [9] and several documents indicate the need to have in mind the phenomenon of accumulation of long-lived radionuclides into the biosphere [10], [11], [12], [13], [14].

3. Basis of the requirement

The detailed explanation of the justification for imposing a collective dose constraint per unit of practice can be found in [3]. Some relevant aspects are introduced in this section, even though neither in the same order nor with the same wording.

As indicated above, the collective dose constraint of per unit of practice was intended to limit the global contribution to the annual doses of all critical groups in the distant future, considering that the continuous discharge of globally dispersed long-lived radionuclides will result in their slow and constant accumulation into the biosphere. In other words, it was pointed out a limitation of the international radiation protection system that can be expressed as follows: In a world with a couple of thousands of nuclear power reactors and a dozen of reprocessing facilities in operation, critical group dose constraints and radiation protection optimization would not prevent the accumulation into the biosphere of extremely long-lived radionuclides⁵.

² Namely, the collective effective dose commitment truncated at 500 years.

³ Although several nuclear reactions produce C-14 in a nuclear power reactor, the most significant is the one with the oxygen of the water, and heavy water reactors have more water per unit of power than light water reactors.

⁴ However, ion exchange resins of the cooling/moderator systems retain certain fraction of C-14 [4], [5], [6].

⁵ This insufficiency was also recognized by the Nordic Countries in 1976 [15].

Regarding the critical group dose constraint, it is technical and economically convenient to assure its fulfillment by retaining radionuclides of relatively short half-life, since those that contribute significantly to the dose have half-lives of up to some decades [7], [16], [17].

On the other hand, the insufficiency of the optimization principle to prevent the accumulation of globally dispersed long-lived radionuclides comes from the uncertainties associated to the use of the concept of the collective effective dose commitment with biological hypotheses, since such commitments only reach significant values as to justify the retention of globally dispersed long-lived radionuclides when extremely low individual doses are integrated in large populations during thousands of years [18], [19], [20]. In this context, it does not seem possible to establish a credible relationship between a calculated reduction of the collective effective dose commitment and the corresponding reduction of the health detriment⁶.

In [3], Dr. Beninson's starting point is an ethical principle: the level of protection of the critical groups in the distant future should be at least the same as today. He also pointed out that such future critical groups would be exposed, in addition to the natural background, to three man-made exposure sources:

- (i) The operating facility,
- (ii) The locally accumulated long-lived radionuclides that will result from the world wide operation, during hundreds of years, of nuclear power reactors and reprocessing facilities, and
- (iii) Other man-made sources not related with the nuclear fuel cycle.

After stating that, and in order to assure the same level of protection, the combined exposures of these sources should not result in a dose higher than the current annual dose limit for chronic exposure of members of the public, i.e. 1 mSv. Then, he postulated the following allocation: 0.3 mSv for direct exposure from the operating facility; 0.5 mSv for other man-made sources and 0.2 mSv for the global contribution or, in other words, the "man-made background", that will be produced by the massive use of nuclear energy. Regarding the last value, he indicates that it can be roughly assumed as equal to the future maximum per caput dose rate in the world population that will arise from the nuclear energy generation practice.

The paper also explains the relationship between the future per caput dose rate in a population and the intensity of the practice, showing that the maximum per caput dose rate in the future can be estimated using, as calculation tool, the collective effective dose commitment without biological hypotheses. If the practice is carried out during a certain number of centuries, the incomplete commitment of the collective dose may be used to that end. Then, assumptions are made about the duration of the practice (500 years), the world population (10^{10} inhabitants) and the expected per caput nuclear energy consumption (5 kWe/man) in order to calculate the collective dose constraint per unit of practice for the nuclear fuel cycle that will restrict the annual contribution to 0.2 mSv. The figure obtained, 40 man Sv/GWey, is further divided between reprocessing facilities (25 man Sv/GWey) and nuclear power reactors (15 man Sv/GWey)

As C-14 is the radionuclide of interest for nuclear power reactors, it should be noted that for this radionuclide the calculation of the collective dose commitment in the world population can be made with a reasonably accuracy because:

- i) Even if the release is not uniform throughout the world, the radionuclide will enter the carbon cycle and will be globally distributed with tendency to homogeneity along the time.
- ii) Any diet of the world population in the distant future will incorporate carbon, since we are a product of its chemistry.
- iii) The per caput dose rate is proportional to the average ratio between C-14 and stable Carbon in the human body.

⁶ In addition to the uncertainties associated with dispersion models, diets and habits in the distant future, it is necessary to take into account uncertainties on the dose-effect relationship at very low doses and the difficulty to estimate the radiological risk per unit of dose because of, for instance, the current cancer mortality and morbidity cannot be assumed as valid in the distant future.

4. Differences between the requirement assumptions and the reality

Firstly, it should be noted certain arbitrariness in the allocation of the above described dose values. For example, values between 0.1 and 0.3 mSv/year for the annual dose constraint applied to direct exposure of the critical group seems to be appropriate and, consequently, values between 0.2 and 0.4 mSv/year may be used for dealing with the global contributions. It seems also difficult to explain why nuclear power reactors have more severe restriction than reprocessing facilities⁷ and it seems hardly justifiable to allocate half of the annual dose limit (0.5 mSv) to other sources of exposure.

Being already in the 21st century, in addition to the previous comment regarding allocation of doses, it is easy to identify that some implicit assumptions did not occur, in particular:

- There was not a strong growth in the world use of nuclear power at the beginning of this century, necessary condition to increase significantly the rate of accumulation into the biosphere of long-lived radionuclides, and
- No country has joined to the effort for retaining C-14, and therefore, the limitation of releases of globally dispersed long-lived radionuclides should be universally implemented to be effective.

It corresponds to emphasize that, quite probably, the absence of international adherence to such initiative is a consequence of the non-occurrence of the expected growth in nuclear energy generation. Actually, such a strong growth is not foreseen even for the next decades [8], [21], [22].

From all the explicit hypotheses in [3], two of them are also relevant regarding the need to retain C-14:

- The estimation of the increase rate of the nuclear electric energy generation in the region and in our country for the following decades; and
- The estimation of the practice intensity in the far future⁸.

After more than three decades, it is clear that the reality of nuclear energy generation is far from the estimation made and, even more, it seems that the estimated generation rate will not be reached also during the next decades.

The second hypothesis has a direct influence in the selection of the collective dose constraint of 40 man Sv/GWey (and indirectly of the value of 15 man Sv/GWey). In effect, these values are directly proportional to the assumption that the contribution of nuclear electric energy in the world will reach (and it will remain) a figure of 5 kWe/man, and recent estimations indicate a value of 1 kWe/man [14]⁹.

Finally, should be stressed that Beninson's paper did not consider that the generation of electrical energy have two effects on the future per caput dose rate due to C-14:

- The use of fossil fuels injects stable carbon into the biosphere, increasing the entire inventory and reducing the C-14/C-stable ratio, with an associated per caput dose rate to world population decrease¹⁰.
- C-14, produced by activation during the operation of nuclear power reactors, would increase the C-14/C-stable ratio and as a consequence, the per caput dose rate to the world population, in this case without practically modifying the entire carbon inventory. This phenomenon corresponds to C-14 and

⁷ If it is taken into account that there will be less reprocessing facilities than nuclear power reactors and that reprocessing is a chemical process, retention seems technically more viable in reprocessing facilities.

⁸ Dr. Beninson affirms that in a few decades, the nuclear electric installed capacity will reach 5 kWe/man in our country and in the region, and that a similar value will be finally reached in the world.

⁹ This reduction might be owing to the efforts that are being done to increase the efficiency in the use of energy [23].

¹⁰ The use of fossil fuels injects stable carbon into the environment and reduces the C-14 concentration. H.E. Suess identified this effect in 1958 and for this reason is called "Suess effect" [24].

does not apply, for example, to I-129 (released principally in reprocessing stage), which is other radionuclide of interest because of its extremely long half-life and global dispersion.

Since several decades, the first effect predominates over the second one and then, the per caput dose rate diminishes.

The C-14/Ctotal ratio began to diminish towards the end of the 19th century, with the beginning of the industrial era. If it were not considered the strong injection of C-14 produced by the atmospheric nuclear tests in the decades of 1950 and 1960, this reduction would be today extremely notable. At present, the ratio is practically equal to that of the pre-industrial era [4], [25]. In addition, current projections indicate that, at least until the year 2030, the fossil fuels consumption will increase at a greater rate than that one foreseen for the nuclear electric energy generation. On the other hand, nuclear energy generation had a lower growth rate than estimations made some decades ago [8], [21], [22].

Then, the need to retain C-14, and the time at which the above-mentioned retention should begin would be a function, principally, of two projections: the expected increase of fossil fuels consumption and the expected increase of nuclear electric energy generation. The time when isotopic equilibrium would be reached (i.e. Suess effect dilution exactly compensates the C-14 injection) seems to exceed at least the next decades. For this reason, perhaps, even though the international concerns remains and the research in this matter continues, there were not international or regional conventions with the target to promote specific actions to retain C-14 in nuclear power reactors.

5. Conclusions

The research seems to confirm that although the international radioprotection system does not contemplate quantitative criteria intended to prevent or to reduce the slow accumulation into the biosphere of extremely long-lived radionuclides, this fact does not still constitute a radiation protection problem.¹¹ Nevertheless, a significant increase of nuclear electric energy generation in the world would show up this omission. In addition, it is stressed that this kind of global problem requires collective actions, for example through international conventions, in order that all the nuclear facilities retain sufficiently radionuclides of extremely long half-lives.

Because of all the above discussed argumentation, the ARN decided to suspend the application of the collective dose constraint per unit of practice for nuclear power reactors. Argentina will follow the world evolution of the nuclear industry in order to be timely prepared to implement this requirement in the framework of international agreements intended to restrict the concentrations of extremely long-lived radionuclides into the biosphere.

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¹¹ Although transboundary migration of radionuclides has already had psychological and even possible economic impacts

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Testing and Intercomparison of Model Predictions of Radionuclide Migration from a Hypothetical Area Source

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Testing and intercomparison of model predictions of radionuclide migration from a hypothetical area source

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Abstract. This work was carried out as part of the International Atomic Energy Agency's EMRAS program. One aim of the work was to develop scenarios for testing computer models designed for simulating radionuclide migration in the environment, and to use these scenarios for testing the models and comparing predictions from different models. This paper presents the results of the development and testing of a hypothetical area source of NORM waste/residue using three complex computer models. There are significant differences in the methods used to model groundwater flow between the models. The hypothetical source was used because of its relative simplicity and because of difficulties encountered in finding comprehensive, well-validated data sets for real sites. The source consisted of a simple repository of uniform thickness, with 1 Bq g⁻¹ of uranium-238 (²³⁸U) (in secular equilibrium with its decay products) distributed uniformly throughout the waste. This approximates real situations,

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such as engineered repositories, waste rock piles, tailings piles and landfills. Specification of the site also included the physical layout, vertical stratigraphic details, soil type for each layer of material, precipitation and runoff details, groundwater flow parameters, and meteorological data. Calculations were carried out with and without a cover layer of clean soil above the waste, for people working and living at different locations relative to the waste. The predictions of the models showed several differences which need more detailed examination. The scenario is available for testing by other modellers. It can also be used as a planning tool for remediation work or for repository design, by changing the scenario parameters and running the models for a range of different inputs. Further development will include applying models to real scenarios and integrating environmental impact assessment methods with the safety assessment tools currently being developed by the IAEA.

KEYWORDS: *NORM, area source, scenario development, model testing, model intercomparison*

1. Introduction

The Environmental Modelling for radiation Safety (EMRAS) project was set up by the IAEA in 2003 to continue and expand upon the work carried out in earlier programs such as BIOMOVs, BIOMASS and VAMP. A working group on materials containing naturally occurring radionuclides (NORM) was set up to look at modelling of the movement of naturally occurring radionuclides in the environment. The aims of this group included looking at legacy sites, existing sites, and planning for future operations, with particular emphasis on the assessment or prediction of health and environmental impacts of NORM. These materials can include waste, residues, and products, with a very wide range of volumes and radionuclide concentrations. The large number of naturally occurring radionuclides and the wide range of chemical and physical properties of these radionuclides have to be taken into account in the models.

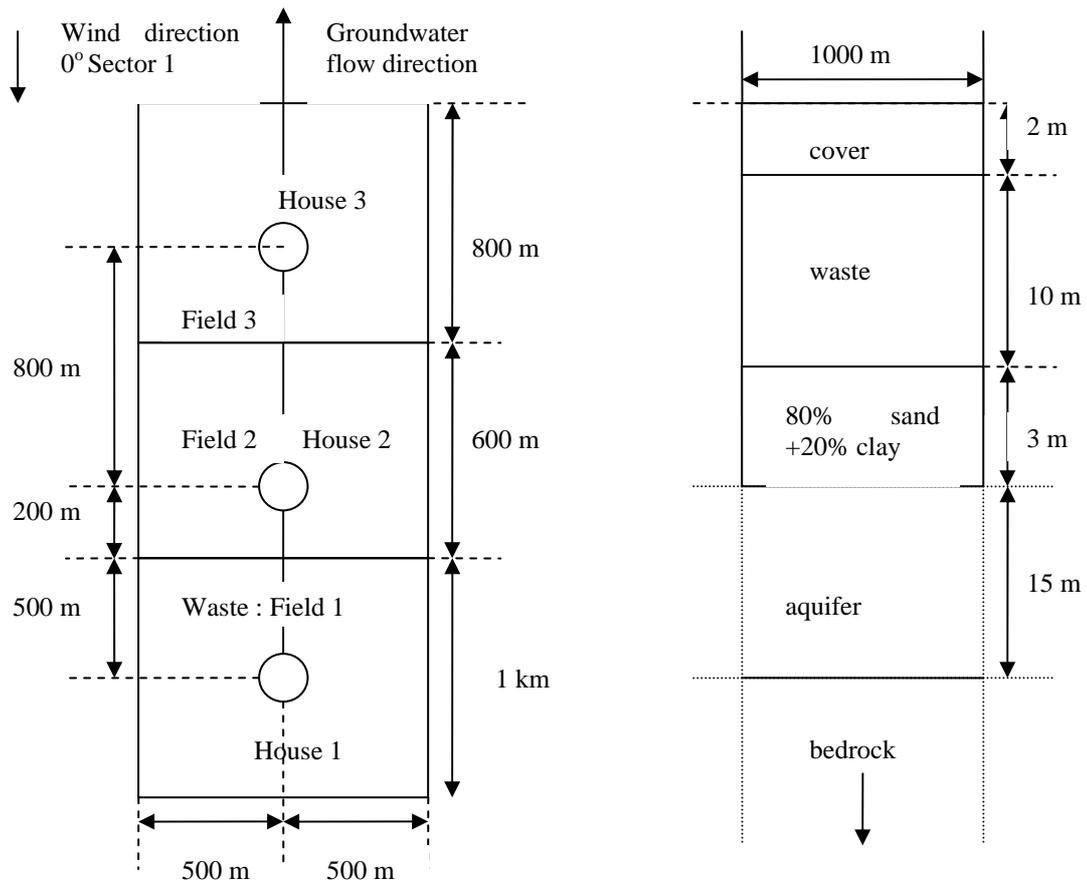
Further aims of the work were to evaluate existing models, develop scenarios for testing and verifying models, and develop new models. Very few models, and no well-documented scenarios for model testing, were available at the start of the program. Therefore a suite of hypothetical scenarios (a point source, an area source, and an area source plus river) was developed for validating and comparing models. This work describes the use of the hypothetical area source scenario for model testing and presents some of the results of the testing and model intercomparisons.

Potential applications of this work include the evaluation of legacy sites, assessment of remediation strategies for legacy sites and existing operations, and prediction of potential impacts of future operations (in particular, assessment of different management strategies).

2. The area source scenario

The main physical features of the site are shown in Figure 1. A detailed description of the scenario, including the modelling input data, can be found in [1]. The radionuclides are assumed to be uniformly distributed throughout the waste. Modellers were also asked to calculate doses to people working in each of the three fields and living in the house within that field. The modellers were asked to carry out the calculations for both covered and uncovered waste, and to consider the effect of rotating the wind rose through all four quadrants.

Figure 1: a plan (left) and vertical profile (right) of the hypothetical site (not to scale)



3. Models used in this work

The models used were PRESTO-CPG ver.4.2, RESRAD-OFFSITE, and DOSDIM + HYDRUS.

PRESTO (Prediction of Radiological Effects Due to Shallow Trench Operations) [2] is a computer package for the maximum annual effective dose to a critical population group from contaminated soil layers, for scenarios involving near surface waste disposal, soil cleanup, agricultural land application, and land reclamation. The package is designed to calculate the maximum annual effective dose to a critical population group for a range of scenarios, and includes models which simulate the transport of radionuclides in air, surface water, and groundwater pathways, and evaluate exposures through ingestion, inhalation, immersion and external exposure pathways. The models used for groundwater dynamics (infiltration, leaching, flow and extraction) are sufficiently detailed to avoid unnecessarily conservative results. The radionuclide migration model assumes the same partition coefficient (K_d) values for parent and daughter radionuclides in decay chains. The model also assumes that the waste is separated from the aquifer by one unsaturated soil layer. Simulation time is limited to 10,000 years.

RESRAD-OFFSITE [3-4] is designed to handle the assessment of both on-site and off-site impacts for situations such as buried waste and landfill (uncovered waste). The package allows for an optional cover layer, up to five partially saturated layers below the waste, and an aquifer (saturated layer). Allowance is also made for the effects of surface water bodies such as ponds, lakes and rivers, and for variations in land use near the disposal site. The models used in the package are complex and require considerable input data, but provide considerable

flexibility when carrying out assessments. The package has been designed to be user friendly and is well documented. The user can select a range of exposure pathways appropriate to the scenario being considered. The ability to carry out both deterministic and probabilistic is also provided. Only deterministic calculations were used for the work described here.

Radionuclides are removed from the waste by leaching, surface runoff, erosion, resuspension (dust) and exhalation (radon). A Gaussian Plume model is used to calculate airborne radionuclide activity concentrations at downwind locations and allows for both wet and dry deposition. The groundwater transport model is also a Gaussian dispersion model, which takes both horizontal and vertical dispersion into account when calculating radionuclide concentrations at off-site locations. This model also considers the decay of the parent radionuclide, the ingrowth of progeny radionuclide(s), and their respective retardation due to sorption/desorption in the solid phase. Drinking water is drawn from a well and the effects of irrigation using water from the well or a surface water body can be estimated using a food chain model.

DOSDIM (+ HYDRUS) calculations involved the use of two modelling packages. DOSDIM (DOSe DIstribution Model) is a compartmental type of model of the biosphere, partly dynamic, depending on the time frame and on the exposure pathways considered. It includes a module, based on a multi-source Gaussian dispersion model, which calculates radon concentrations in the air from point and area sources [5]. The DOSDIM package has been used in the BIOMOVS [6], VAMP [7-8] and BIOMASS [9] model validation and verification studies.

The radon dispersion was calculated by subdividing the area of the repository (1 km x 1 km) into of 100m × 100m cells. At the point of interest (e.g. house 2 in Figure 1), the contribution from each cell was calculated, and the individual contributions were then summed to give the total radon concentration [1]. For the covered waste scenario a radon exhalation rate of 0.02 Bq m⁻² s⁻¹ was used, while a value of 1 Bq m⁻² s⁻¹ was used for the uncovered waste scenario. For the uncovered waste situation the exhalation of radon from the waste directly into the house was included. The effects of erosion on the thickness of the cover were also allowed for in the calculations.

HYDRUS 1D and HYDRUS 2D were used for modelling the transport of the radionuclides in the variably saturated medium under the waste. Both models use finite element methods to simulate water and solute movement in unsaturated, partially saturated, or fully saturated porous media. For this work the one-dimensional version HYDRUS 1D was used to model the transport of the radionuclides through the vadose (unsaturated) zone under the waste into the aquifer (saturated zone). The concentrations of the radionuclides in the aquifer at the location of the exposure point (a well at the house) were then calculated with HYDRUS 2D using the output values from HYDRUS 1D as input.

4. Results

A number of exposure pathways were considered, including external exposure, inhalation of radon and contaminated dust, and ingestion of contaminated water and food. The effects of erosion on covering material were also considered, and the RESRAD-OFFSITE package gave the modellers the option of considering the effects of different types of land use following the deposition of the waste material.

The important factors which can influence the impact of surface and near-surface disposal of waste include future land use, the diet and general life-style of the inhabitants of the area, rainfall, groundwater transport, surface water transport, radon exhalation, the presence or absence of covering material above the waste, and erosion. The models should be able to simulate each of these processes.

Future land use can both influence and depend on the outcome of an assessment of the potential impact of a proposed waste repository, and can also affect doses once the waste is emplaced.

There are a large number of exposure pathways to be considered for scenarios of the type discussed here. However, in general, the exposure pathways can be put into groups; these groups are external exposure, exposure to airborne radionuclides (dust, radon, and thoron), food chain pathways, surface water transport pathways and groundwater transport pathways.

The effects of surface water transport were not considered in this scenario. Covering the waste should significantly reduce the contributions of airborne radionuclides, and external exposure, provided the cover remains intact. However, for long-term predictions the effect of erosion can be expected to be important. Erosion of the cover should increase the dose contributions from the airborne radionuclides and from external exposure, while erosion of the waste material itself should decrease the contributions from all exposure pathways. In addition, because groundwater transport is much slower than airborne transport, there should be significant delays in the effects of the groundwater pathways relative to the placement of the waste and the removal of the cover and waste by erosion. These considerations helped to guide the modellers in both developing the models used in this study, and in interpreting the results of the scenario testing.

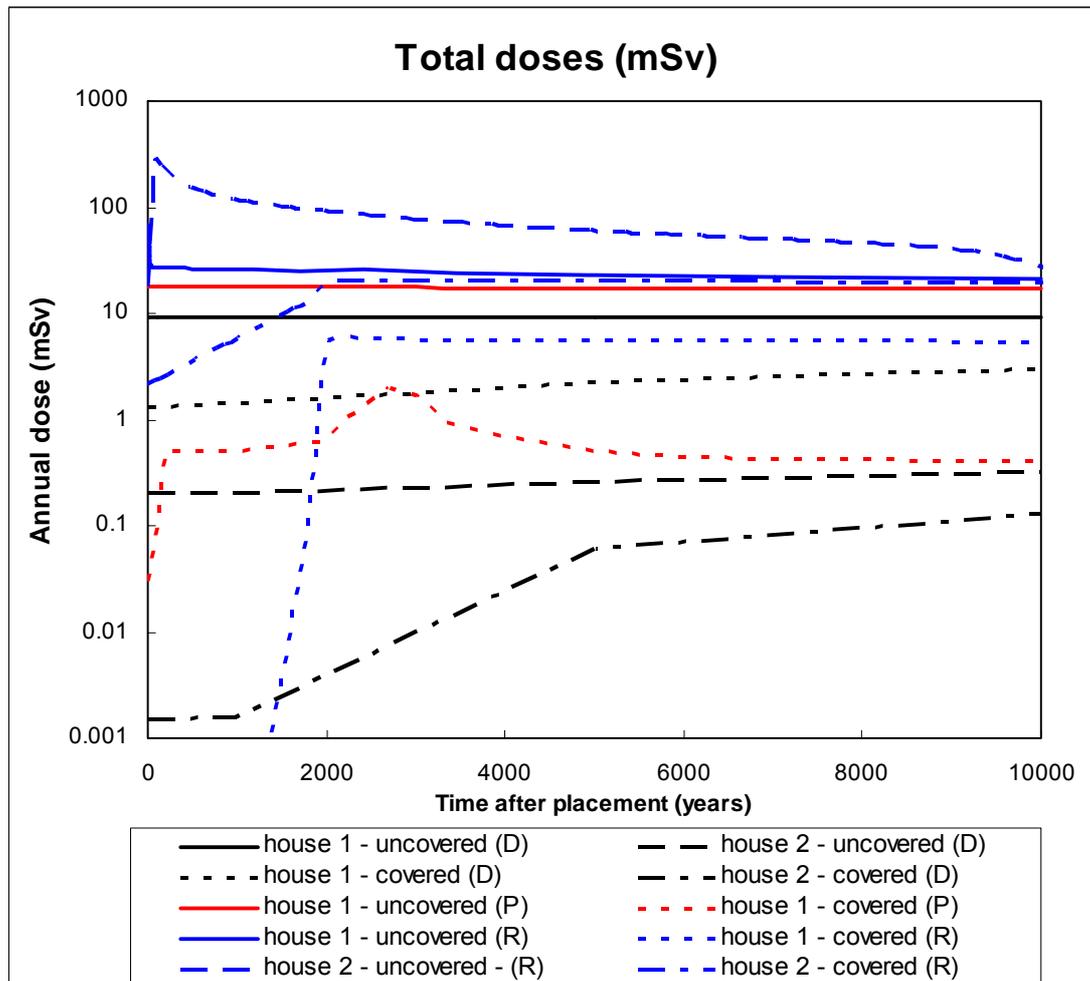
The detailed results are presented in the main report [1] of the NORM working group (to be published). Some of these results are presented in the following sections. Total dose predictions are discussed in Section 4.1, dose contributions from individual exposure pathways are discussed in Section 4.2, and ^{238}U concentrations in well water are discussed in Section 4.3.

Most of the results presented in the following sections were generated by the RESRAD-OFFSITE and DOSDIM (+HYDRUS) models.

4.1 Total doses

Total doses were calculated in all models by summing the contributions from all exposure pathways included in the model. The results for house 1 and house 2 are shown in Figure 2.

Figure 2: Total doses for houses 1 and 2, as calculated by the models PRESTO, DOSDIM and RESRAD-OFFSITE.



In Figure 2 the labels (D), (P) and (R) refer to the models DOSDIM (+ HYDRUS), PRESTO and RESRAD-OFFSITE respectively. DOSIM results are presented as black lines, while PRESTO results are represented by red lines and RESRAD-OFFSITE results are represented by blue lines. This allows comparison of the prediction of different models by comparing curves of the same line type but with different colours. The solid lines refer to the doses for a resident of house 1 for uncovered waste. The dashed lines (---) refer to house 2 for uncovered waste. The dotted lines (.....) refer to house 1 for covered waste. The broken dashed lines (- - -) refer to house 2 for covered waste.

Figure 2 shows a number of interesting features. In general, the RESRAD-OFFSITE predictions are higher than those from PRESTO and DOSIM (+HYDRUS). In particular, the RESRAD-OFFSITE results appear to be a factor of approximately 100 higher than the DOSDIM (+ HYDRUS) results. The ratio between the waste with no cover case and the waste with cover case is approximately the same for times greater than 5,000 years, but is greater for RESRAD-OFFSITE than for DOSDIM(+ HYDRUS) for times less than 5,000 years. The reason(s) for these differences are not understood at this time.

In RESRAD-OFFSITE the erosion rate was calculated from the rainfall, infiltration and run-off data. For this scenario the erosion rate was estimated as 0.99 millimetres per year

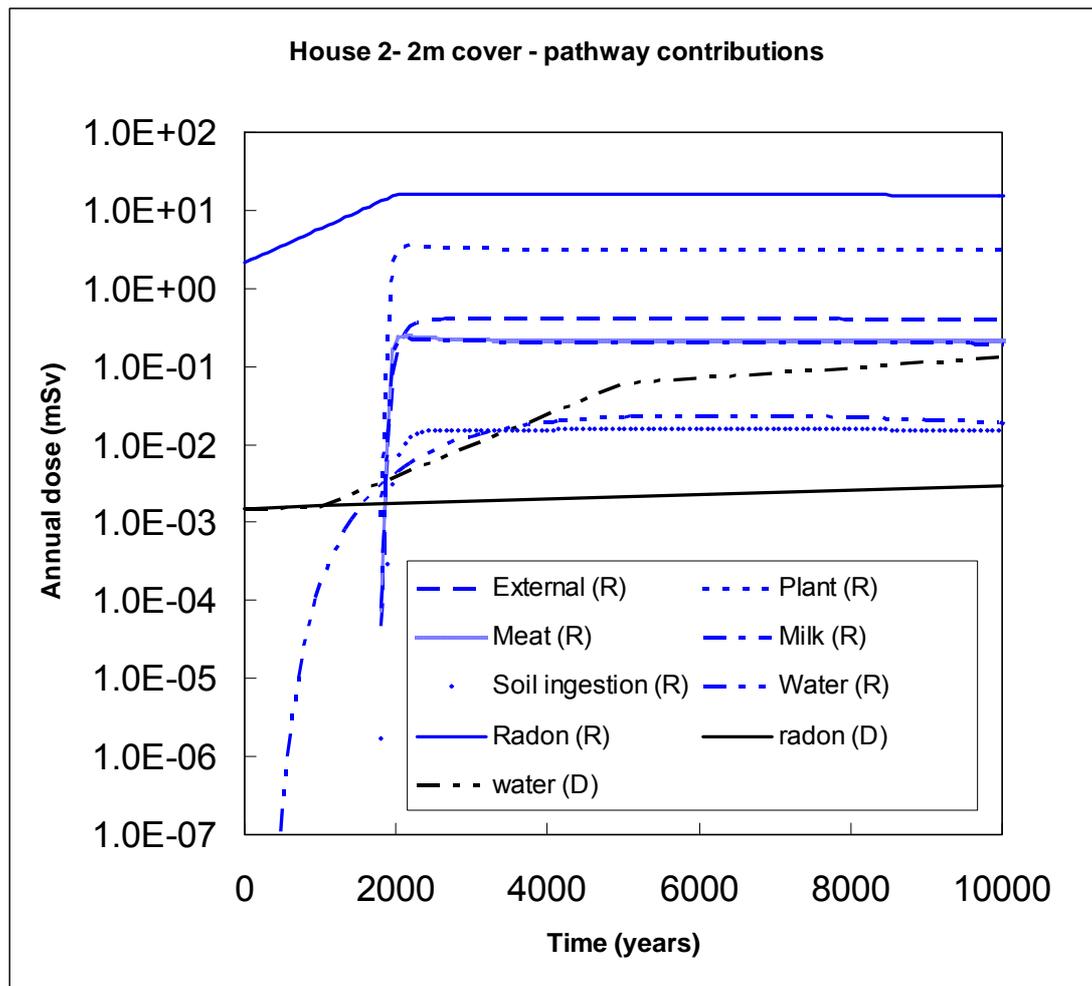
(approximately 1 metre per 1,000 years). The erosion of the cover could explain the increase in dose over the first 2,000 years for the covered waste.

To examine the reasons for the reasons for this increase in dose, it is necessary to look at the contributions to the total dose from individual pathways.

4.2 Dose contributions from individual pathways

The predicted dose contributions for the important exposure pathways are presented in Figure 3.

Figure 3: Dose contribution from individual pathways, calculated by RESRAD-OFFSITE



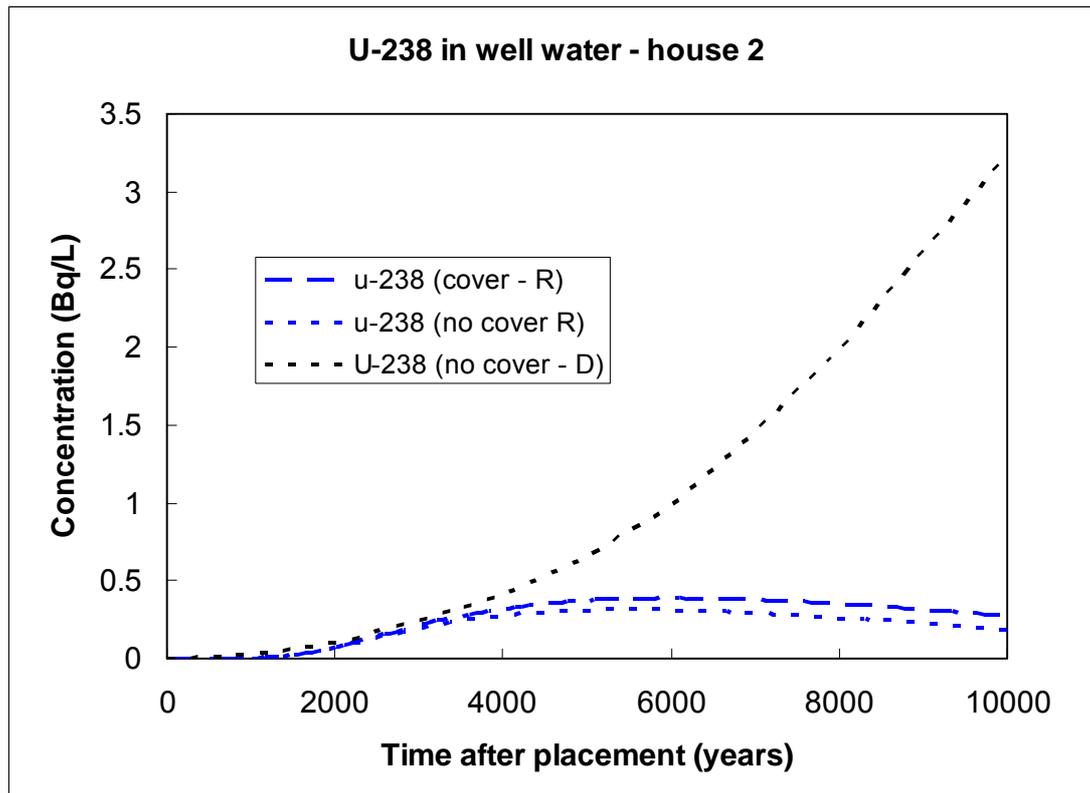
In Figure 3 there are several points to note. RESRAD-OFFSITE predicts significantly higher radon doses than DOSDIM (+ HYDRUS). The reason for this difference is not understood at this time. Leaving aside the possibility of mistakes by the modellers, there are several factors that can influence indoor and outdoor radon concentrations. Outdoor radon concentrations are strongly influenced by atmospheric conditions, and radon concentrations inside a dwelling tend to be dominated by the building materials used in constructing the dwelling, the ventilation rate, the rate at which radon enters the dwelling as a result of exhalation from the soil under the dwelling, and the presence or absence of a basement. Clearly, further work is needed to explain the differences in the model predictions.

Figure 3 indicates that the increase in total dose predicted by RESRAD-OFFSITE during the first 2,000 years is due to the increased exhalation of radon as the cover is removed by erosion. This behaviour is not reproduced by the other two models.

4.3 Dose contributions from individual radionuclides

Another useful comparison of model predictions was achieved by examining the predicted radionuclide concentrations in well water. The results of these calculations for RESRAD-OFFSITE and DOSDIM (+ HYDRUS) are shown in Figure 4.

Figure 4: ^{238}U concentrations in well water for house 2.



DOSDIM (+HYDRUS) predicted the same concentrations for both the uncovered and covered waste cases. Both models predicted that the ^{238}U will start to appear in the well water after approximately 1,000 years. However, beyond that point the model predictions differed markedly.

If there were no erosion the ^{238}U concentration in well water should be expected to increase with time until most of the ^{238}U has been leached from the waste. However, erosion should remove the waste and hence reduce the source term. Therefore the ^{238}U concentration should increase at first but then start to decrease as the effects of erosion reduce the total amount of ^{238}U (in the waste) available for leaching into the groundwater. The effect of cover material would be expected to delay the expected decrease in ^{238}U concentration in well water because of the extra time needed to erode the cover. The RESRAD-OFFSITE predictions are consistent with this.

5. Conclusions

There appear to be significant differences between the predictions of the three models tested in this study. There are many processes which can affect the health and environmental impacts of surface and near-surface disposal of waste. These include erosion (of both cover

material and the waste itself), leaching, ground water transport processes, radon exhalation, rainfall, meteorology, future land use, and the use of (possibly) contaminated water for irrigation and domestic use. These processes are complex, which means that the development of models for simulating the transport of radionuclides in the environment for this type of scenario is not a simple matter. In addition, because of the complexity of the models and the large amount of input data needed by the models, there are many choices to be made by the user when setting up computer packages to model this type of scenario. This was noted by the DOSDIM user, who pointed out several important omissions in the original scenario specifications, and by the RESRAD users, who had difficulty agreeing on the land use specifications for the calculations. The resolution of these difficulties led to several important conclusions:

1. it is not always possible to specify the scenario without going through an iterative process of testing and modification;
2. good communication between modellers is essential, to ensure that all modellers use the same site specifications and the same values for environmental parameters, and produce results that can be directly compared.
3. there appear to be some significant differences between the predictions of different models.

There are significant differences between the PRESTO and RESRAD-OFFSITE results. There also appear to be significant differences between the predictions of the DOSDIM (+ HYDRUS) and RESRAD-OFFSITE, models particularly with respect to the contribution from radon inhalation to the total doses predicted by these models, and the effect of cover above the waste. More work is needed to explain and resolve these differences.

A program to follow-up the EMRAS work has been suggested. Clearly, with respect to NORM there are several issues that still need to be addressed. These include the development of more real scenarios for model testing, further development and testing of the models which are currently available, and the development and testing of models to look at scenarios (such as tailings dams and lakes) which were not looked at in the EMRAS program

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Initial Basis for Agronomic Countermeasure Selection Following a Nuclear Accident

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Initial Basis for Agronomic Countermeasure Selection Following a Nuclear Accident

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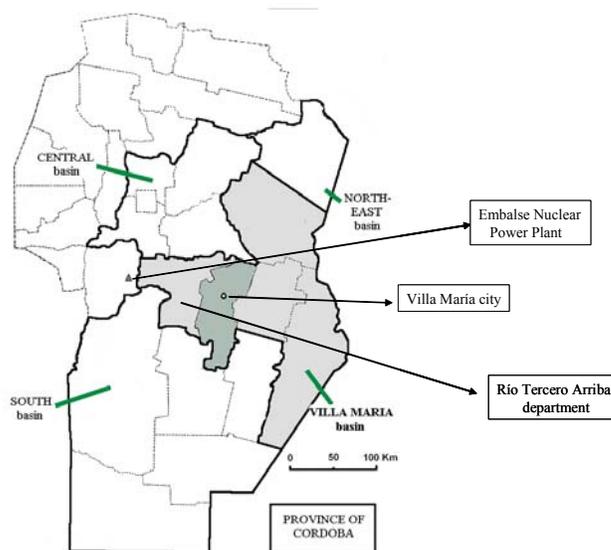
Abstract. During the recovery stage, following a nuclear accident, application of agricultural countermeasures will be relevant to the minimization of the radiation induced detriment due to ingestion of locally produced contaminated foodstuff, as long as the magnitude of the averted dose is sufficient to justify their implementation. Nuclear emergency planning in Argentina currently holds food ban as the accepted countermeasure, at least until other measures are taken. Though it may ensure no residual collective dose, food ban may also imply very high costs, compared to other alternatives, specially due to the need of disposing off perishable food such as milk. Therefore, an exhaustive evaluation of all the alternatives, considering both quantitative and qualitative factors is still needed to identify optimal countermeasure strategies, bearing in mind also that decisions made during the early phase of an emergency will affect the fate of the measures to be taken later. As a first step in this direction, a basic quantitative decision-aiding technique, the cost-benefit analysis, is carried out for comparison of countermeasures related to Cesium contaminated cow-milk which are considered feasible for implementation in Argentina. Countermeasures total costs are estimated from various local sources, while their effectiveness are adopted from international bibliography. At this stage, a simple theoretical example considering milk contamination in the surroundings of the Embalse Nuclear Power Plant is used for a generic analysis, since actual collective doses and costs can only be calculated for a specific modelled scenario.

KEYWORDS: *Agronomic countermeasures, food contamination, ¹³⁷Cs in milk, cost-benefit analysis*

1. Introduction

The Embalse Nuclear Power Plant (CNE), is a CANDU type reactor set in the city of Embalse Rio Tercero, 115 Km ONO of Villa María city. Villa María is the capital city of General San Martín department (Pop. 120000), geographical centre of the Villa María milk basin. This basin comprises also (among others) the Río Tercero Arriba department to the west, which neighbours the city of Embalse (Fig. 1).

Figure 1: Setting of Embalse Nuclear Power Plant in the Argentinean province of Cordoba



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Villa María basin represents 50% of the (cow) milk production of the Argentina province of Córdoba, which in turn contributes with approximately 34% to the 9000 million litres produced annually in the country [1]. Although CNE is not strictly placed within the basin, its closeness to such a big milk production area raises concerns about the risk of milk contamination in case of a nuclear accident. Experience has confirmed how relevant the contribution of ingestion of ^{137}Cs contaminated milk can be to the total dose during the mid and long term periods after a nuclear accident, and how important agronomic countermeasures are in significantly reducing such contribution, specially if applied as early as possible [2]. The level of food contamination will ultimately depend on site specific variables, some of which can be quantified beforehand (e.g. transference factors) and some of which can be quantified exclusively after the accident (e.g. soil deposition). Based on this, several countries have developed complex modelling programs for emergency preparedness, capable of estimating, as early as possible, radionuclide deposition over an area after an accidental release, deposited radionuclide transfer through different pathways to man for dose calculation, and capable also of helping determine the most adequate countermeasure strategy for reducing such doses [3, 4].

In Argentina, norms dictated by the Nuclear Regulatory Authority (ARN) mention only food ban as the protective action to be taken [5] at a generically optimized intervention level (IL) of 1000 Bq/L [6]. Milk ban implies its withdrawal from commercialization markets and its replacement with alternate supplies. While this assures minimal risk to consumers, being milk a perishable product, it will go waste if no other complementary action (e.g. cheese manufacturing) is taken. If no replacement food is available, milk ban may not be justified at that IL. Moreover, this action could result very cost-ineffective [7], particularly in an area where high volumes may be involved. If other protective actions (agronomic countermeasures) were applicable and able to lower ^{137}Cs concentration in milk below IL, then a comparison should be made in order to select the optimal intervention strategy, more likely for the long term after the accident.

Despite some discussions on the topic, no agronomic countermeasure directed to limit the transfer of radionuclide in the food chain has been thoroughly studied for milk contamination with ^{137}Cs after a nuclear accident in Argentina, as part of emergency preparedness plans. Neither is any specific model available yet, however simple, which can provide preliminary estimations of food contamination, as an aid in the selection of appropriate agronomic countermeasure strategies, when an accident has already occurred and pressure on decision makers is high.

The objectives of this work were: a) to perform an empirical calculation of milk activity concentration and consequent projected collective dose, for a hypothetical ^{137}Cs release from CNE, and b) to introduce a discussion on the applicability of some agronomic countermeasures within this scenario, while performing a preliminary selection of an optimal countermeasure strategy by means of a simple cost-benefit analysis.

2. Materials and Methods

2.1 Villa Maria Basin Scenario

The area of concern in this work is the Villa María milk basin (VMb), in particular the portion comprised by the departments of General San Martín (GSM) and Río Tercero Arriba (RTA) (see fig.1).

As required for this work, farmland area statistics were available for GSM only [8], while production statistics were available averaged for the whole of VMb [9]: Of the approximately 500000 ha total department area, about 37% is covered by dairy establishments, which have an average area under fodder of 70% of total farmland (approximately 52% alfalfa (*Medicago sativa* L.) based permanent pastures and 18% annual fodder crops like oats and maize for silage). Maize silage and hay made of pasture spring/summer surpluses and, occasionally, of summer fodder crops, are used for compensating the decrease in pasture productivity during the colder months. Winter fodder crops are directly consumed by the dairy cows. For the sake of simplicity, since depending on the date of occurrence of a hypothetical accident any of the fodder alternatives may be contaminated, total fodder area will be considered for the remaining of this work as consisting only of alfalfa pastures. Approximately 57% of total farmland area is destined to support dairy cows (e.g. milkers and dry cows). Considering an average ratio of 1.09 dairy cows / ha, there are approximately 114940 of them

in the department, 78% of which are milkers, with a daily average yield of 15.6 L/d. Consumption of fodder in the basin was estimated at 12,3 kg (dry weight)/d.

Following an example from a case study of agronomic countermeasure analysis for a hypothetical nuclear accident in northern England, available from the STRATEGY project [7], a 25 km² site was selected as the ¹³⁷Cs deposition area following a hypothetical accidental release from CNE. Unlike the example, lack of detailed information on a scale lower than that of General San Martin department, required the site to be an imaginary one, with productive statistics resembling those of the departments, as shown in Table 1. This site will be used for quantification of collective doses and countermeasure costs.

Table 1: Productive statistics for the 25 km² site affected by ¹³⁷Cs deposition (values being a scaled down reproduction of those of averaged for General San Martin department).

Total affected area	2500 ha (25 km ²)
Total affected fodder area	647 ha
Dairy cows	574
Milkers	448
Milk yield	15,6 L/d
Fodder consumption rate	12,3 kg (dry weight)/d

2.2 Activity Concentration in Milk and Collective Dose Calculations

To perform a cost-benefit analysis for countermeasure comparison, the expected activity concentration in milk, as well as the projected, averted and residual collective doses need to be calculated. The equations used for these calculations were derived from [7, 10]:

$$A_m(t) = DEP \cdot T_{ag} \cdot FR \cdot TF \cdot e^{-\ln 2 \cdot t / T_{eff, ecol}} \quad (1)$$

where

- $A_m(t)$ = Activity concentration in milk (Bq/L) at time t.
- DEP = Initial deposition (Bq/m²).
- T_{ag} = Aggregated transfer factor from soil to plant (m²/kg).
- FR = Feeding rate of animal (kg/t).
- TF = Transfer factor from feed to animal.
- $T_{eff, ecol}$ = Effective ecological half-time for, in this case, ¹³⁷Cs.
- t = Time since deposition.

Activity concentration in milk was calculated monthly, due to seasonal variations in pasture production and consumption, while doses were calculated for a whole year:

$$PD_{coll} = \sum (A_m \cdot Y_a) \cdot N^o \cdot DCF_{ing} \quad (2)$$

where

- PD_{coll} = Projected collective dose (milk ingestion) in absence of countermeasures (person Sv).
- A_m = Monthly average activity concentration of ¹³⁷Cs in milk (Bq/L).
- Y_a = Monthly animal yield of milk (L/m).
- N^o = Number of animals considered.
- DCF_{ing} = ¹³⁷Cs dose conversion factor for ingestion (Sv/Bq).

$$AD_{coll} = PD_{coll} \cdot E_{CM} \quad (3)$$

where

AD_{coll} = Averted collective dose due to countermeasure application (person Sv).
 PD_{coll} = Projected collective dose (milk ingestion) in absence of countermeasures (person Sv).
 E_{CM} = Countermeasure effectiveness (dimensionless).

$$RD_{coll} = PD_{coll} - AD_{coll} \quad (4)$$

Where

RD_{coll} = Residual collective dose (person Sv).
 PD_{coll} = Projected collective dose (person Sv).
 AD_{coll} = Averted collective dose (person Sv).

2.3 Cost-benefit Analysis

For this work, a simple cost-benefit analysis was used for countermeasure comparison, as described by the ICRP [11]. It involves calculation of a total cost, represented by the sum of the monetary cost of implementing the countermeasure plus the cost of detriment, obtained from transforming the residual collective dose into monetary value. The cost of a collective dose unit (person Sv) in Argentina is US\$ 10000 [5].

Countermeasure application costs were derived from information provided by different service and supply providers, except for the use of the Cs-binder ammonium hexacyanoferrate (ACFC), which is not produced in the country and is rarely imported, so its cost was approximated from bibliography [12]. May 27, 2008 exchange rates were used (US\$ 1 = Argentine \$ 3,15 = € 0,63). Implementation of countermeasures and their costs were discussed also with some farmers. The costs of managing wastes that may be produced by the individual countermeasures were not included in the cost-benefit analysis, nor were incremental doses to those implementing them taken into account.

3. Results and Discussion

3.1 Radiocaesium Deposition Simulation and Transfer to Milk Calculation

An accidental emission of $1,38 \cdot 10^{15}$ Bq ^{137}Cs from CNE was simulated using ARN SEDA and LLNL Hotspot codes. Parameters used were a 3 hour release after a 2 hour delay time, at an altitude of 10m and a $6,6 \cdot 10^4$ cal/s heat flux.. Wind speed was 4 m/s, blowing from the west, and class D stability.

The simulation showed an area covering slightly over 25 km² and with a ground deposition of approximately 390000 Bq¹³⁷Cs/m² (similar to that studied at the STRATEGY example), situated some 30 km to the east of CNE, near RTA's western limits and 85 km short of Villa María city. The geographical centre of VMb received a simulated deposition of approximately 75000 Bq¹³⁷Cs/m², within a 455 km² area receiving 50000-100000 Bq¹³⁷Cs/m².

Equation (1) was used to calculate the monthly concentration activity of ^{137}Cs in milk produced at the site receiving 390000 Bq/m². Since this work is concerned only with radiocaesium transfer to milk, the first two months after the hypothetical accident were not considered, being iodine isotopes the relevant radionuclides during that period. On the other hand, direct deposition on plant surfaces has been found not to be of concern after approximately that lapse of time [13], so it was not included in the calculation. The accident was simulated to occur in September, so that calculations would begin with the season of highest pasture productivity, when pasture feeding rate was assumed to be 12 Kg dry weight/day, followed by 5 kg/day in autumn, 2 kg/day in winter and 10 kg day in spring. Results are presented in Table 2.

Table 2: Calculated ^{137}Cs concentration activity in milk after a simulated 390 KBq/m² soil deposition.

Month	^{137}Cs concentration activity (Bq/L)
December	295,7
January	279,1
February	267,5
March	103,6
April	97,8
May	92,3
June	34,8
July	32,9
August	31,1
September	146,6
October	138,3
November	130,6

The VMb scenario did not produce ^{137}Cs contaminated milk above the IL of 1000 Bq/L in any of the twelve months (following the first two month period) after the simulated event. Activity concentration values were up to one order of magnitude lower than those presented in the STRATEGY Project example. This difference was mainly attributed to the aggregated transfer factor from soil to plant. Differences of several orders of magnitude have been found for Cs transfer to plants depending on soil type. A generic classification based on soil texture (peat, sand, loam, clay, in decreasing order of availability to plants) is oftently used to reflect such variability. The main process controlling root uptake of Cs is the interaction between the soil matrix and the soil solution containing that Cs, which depends on the cation exchange capacity of soils. Higher clay content correlates positively with the cation exchange capacity and with competing cations like potassium [13]. Productive soils in GSM and RTA can be classified mainly as Haplustols, with a clay loam texture and pH slightly above 7 [14]. Main transfer factors references [15, 16] provided values for perennial pastures/fodder, but were discarded for this work as they referred to equilibrium conditions, which are reached after some years of interaction between the soil matrix and the Cs in the soil solution, decreasing plant uptake. Two other references were considered more appropriate [13, 17], as they referred to transfer factors for use following accidental emissions. Both provided values three orders of magnitude higher than those at equilibrium for a clay/loam/high nutrient status type of soil like Haplustols, although they were calculated only for cereals and for natural grasses. The higher T_{ag} value of $8 \cdot 10^{-3} \text{ m}^2/\text{kg}$ was used, derived from curves for natural grasses in a Chernozem soil after the Chernobyl accident [13]. Only one published ^{137}Cs transfer factor was found for an argentine soil [18], but not enough information was available, appart from the use of a clay loam type of soil, to use it unhesitatingly. Anyway, it provided slightly over half the value finally used in this work. Accordingly, Effective ecological half-time value of 1 year was derived from the same reference [13]. Cs transference factor from feed to milk (value $7.9 \cdot 10^{-3} \text{ d/L}$) was taken from IAEA/IUR [15] and dose conversion factor for ^{137}Cs ingestion ($1,3 \cdot 10^{-8} \text{ Sv/Bq}$), was taken from ICRP [19].

Even though one of the highest T_{ag} values available in the literature for a type of soil similar to that of VMb was used, based on ARN norms, no protective action would be justified for this scenario. Collective Projected Dose, as calculated by equation (2), was 5,8 person Sv for the twelve months period. The simulated event would be of even less consequence nearer to the main productive area (as far as internal doses due to ^{137}Cs ingestion through contaminated milk goes) for the first year.

3.2 Cost-benefit Analysis and Countermeasure Discussion

A detailed description of currently feasible agronomic countermeasures for radionuclide contamination of food, including their effectiveness and some costs, elaborated by the STRATEGY Project, is available also at the IAEA web page [20]. Some of these countermeasures have been extensively implemented in a real scenario following Chernobyl accident [13]. Based on those references, and taking into consideration local productive characteristics, six countermeasures (including milk ban) were considered eligible to perform a quantitative comparison by means of a simple cost-benefit analysis, for the simulated VMb scenario. It needs to be remarked that, as

suggested by ICRP, final decision on protective action application will depend on less quantifiable factors such as, among several others, public anxiety due to either implementation or lack of implementation of actions [21]. Thus, it may well happen that, depending on the magnitude of the event and on costs, it is preferable to apply a countermeasure which will lower radionuclide activity concentration in milk below already acceptable levels, if by so doing local and/or foreign consumer assurance is gained.

The analysed countermeasures are presented in Table 3, together with their application costs and effectiveness.

Table 3: Relevant countermeasure characteristics and results of the cost-benefit analysis.

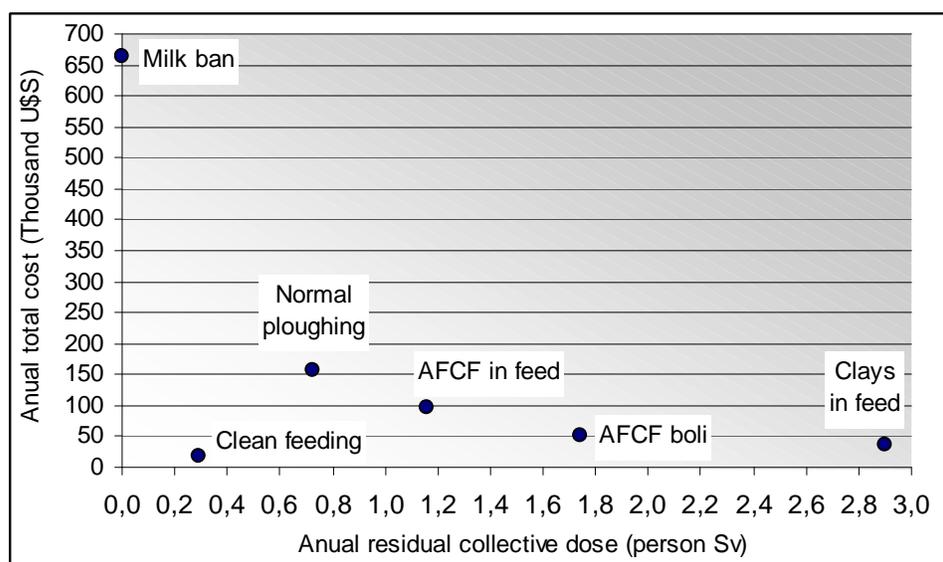
Total collective dose (person Sv)	Countermeasure	Cost (US\$)	Effectiveness	Residual collective dose	Total cost (US\$)
5,8	Normal ploughing	150311	0.50 (0.88) ^a	0,7	157561
5,8	Clays in feed	7598	0.50	2,9	36598
5,8	AFCF in feed	83804	0.80	1,2	95404
5,8	AFCF boli	32752	0.70	1,7	50152
5,8	Clean feeding	16174	0.95	0,3	19074
5,8	Milk ban	663237	1.00	0,0	663237

^a Higher effectiveness due to clean feeding implementation for 10 months prior to full production of new pasture, was used.

The analysis was performed following equations (3) and (4), out of a total collective dose of 5.8 person Sv. The total cost, shown in Table 3, represents a figure of merit, in the way that the countermeasure with its lowest value will be the best choice, provided it guarantees ¹³⁷Cs activity level in milk below optimized IL.

As can be seen on figure 2, clean feeding is, as far as this work goes, the best choice (lowest annual total cost) for a protective action to be applied when facing the VMb scenario.

Figure 2: Cost-benefit plot for the six countermeasures discussed in the work.



Clean feeding implies substitution of the whole daily ration (consisting, on average, on 12 kg dw fodder + 4 kg dw maize grain), for all the dairy cows in the affected area, for as long as the action is needed. Fodder is almost always produced within a dairy establishment, so substitutes will have to be purchased from the nearest uncontaminated areas, as available. Since maize silage is produced in

Argentina only for self use, hay rolls (preferably of alfalfa or oats) will be the only alternative. Concentrate (maize grain) was also included in clean feeding costs, although it is normally purchased by the farmer. Approximately 70 g of a mineral/vitamin supplement should be added to the ration, especially when fresh pasture cannot be provided. Clean feeding resulted in a notoriously cheap option compared to the other countermeasures (except for use of clays in the feed), and its implementation does not imply significant extra activity for the farmers, other than a new schedule for ration distribution and preventing cattle consuming of contaminated pastures. It also assures the second highest averted dose, rendering the lowest total cost. The main setback for its application would be the lack of available substitutes, specially if large areas are affected. Also costs would increase with increasing demand on those substitutes.

The second choice is distribution of clays (bentonites were considered) with the ration. It is a very simple option, which only requires purchase and transportation of large amounts of clay minerals, since applications of up to 1g/kg body weight per day are recommended. Normally, after grazing on pastures, dairy cows are fed the concentrate (and supplementary fodder like silage or hay, in winter) at least twice a day when milked. It would be better to incorporate clay into pelleted ration, however, balanced rations are not normally provided on average establishments. It is also the cheapest option of all (clay minerals are locally extracted), but has the lowest effectiveness. In our simulation, residual dose was not a problem, but in an improbable worst case scenario with ^{137}Cs activity in milk found to be, for example, thrice the IL, this countermeasure would not be an option, however simple and cheap. Use of AFCF in intraruminal boli and mixed with the feed were the third and fourth choice, respectively. They are much effective, though not as cheap as clay mixing. AFCF boli were considered the most cost-effective countermeasure in the STRATEGY Project example. However, they need to be imported, with high uncertainties regarding what their actual price will be if really needed. Boli are not even a commercial product, and fabrication techniques should be learnt here first.

Within this work, Normal ploughing refers to the combine actions of ploughing at a depth of 20 cm, soil refining, sowing of new pasture and fertilizing (nitrogen + Phosphorus). Argentinean soils are generally well provided of potassium, so it is normally not included. Ploughs are available in almost any farm, and farmers are well used to their utilization. Deeper ploughing is not favoured by producers who do not want the topsoil mixed with deeper (less quality) soil. This countermeasure represents a particular case of protective action, since it also helps reducing external doses from contaminated ground (not accounted for in the cost-benefit analysis). Also, it is the only countermeasure among the other selected within this work which does not need to be implemented every year, so that it would show a lower total cost in a cost-benefit analysis performed for a 5 year period. It should be noted, however, that ploughing and reseeded implies a 10 month grazing exclusion period until full production of alfalfa is regained, during which clean feeding has to be implemented, thus increasing this countermeasure effectiveness from an original 0.50 to 0.88 for the first year of application only.

The only cost considered in this work for milk ban (and disposing of the milk) was the value paid to the farmer for each produced litre. It resulted in the most expensive, by far, of the studied countermeasures. Although its 100% effectiveness was of no special significance in a low milk contamination situation like the one simulated in this work, its very high cost indicate that being clean feeding or Cs-binders use feasible, milk ban would still not be a choice action when IL is exceeded.

4. Conclusions

In case of a ground deposition of 390000 Bq/m^2 within the milk production areas to the west of the Embalse nuclear power plant, following an accidental release of ^{137}Cs , concentration activity in milk would not exceed intervention levels of $1000 \text{ Bq}^{137}\text{Cs/L}$, after a short term period of approximately two months. This would be due to the apparently low transference of Cs from the type of soil most common in the area to the plants conforming the pastures in which local milk production is based. Site specific aggregated transfer factors from soil to plant should be obtained, either through semi-mechanistic models, or through experiments using local soil, for more accuracy. In case countermeasure implementation was required, clean feeding of dairy cattle to prevent milk contamination appears to be the first choice, due to its low monetary costs, high averted dose and easy implementation. Banning and disposal of contaminated milk was found to be highly expensive.

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Analysis of Generic Exemption Levels for Radioactive Material

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“Analysis of Generic Exemption Levels for Radioactive Material”

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Abstract. In essence, exemption may be considered a generic authorization granted by the regulatory body, which, once issued, releases the practice or source from the requirements that would otherwise apply, in particular, the requirements relating to notification and authorization. The exemption figures included in the Basic Safety Standards BSS 115 were derived from three scenarios postulated in the document “Radiation Protection 65” of the Commission of the European Communities considering quantitative exemption criteria. This paper describes and analyses these scenarios and assesses the degree of conservatism of the parameters involved in each exposure pathway. The scenarios contemplate Normal Use (workplace), Accidental (workplace) and Disposal (public) covering external exposure, ingestion and inhalation. These scenarios were used to calculate both exempt activity concentrations (Bq/g) and total activity (Bq).

KEYWORDS: *Exemption, Generic Exemption Levels.*

1. Introduction

Exemption allows regulators to release practices or sources from compliance with specified requirements if they judge that the application of such requirements (e.g. notification, authorization) are unwarranted.

The concept of exemption has been in international use for some years and it was already included in the Commission’s 1990 recommendations (ICRP 60 [1]), as follows: ...“In order to avoid excessive regulatory procedures, most regulatory systems include provisions for granting exemptions. The Commission believes that the exemption of sources is an important component of the regulatory functions. There are two grounds for exempting a source or an environmental situation from regulatory control. One is that the source gives rise to small individual doses and small collective doses in both radioactive materials containing radionuclides of natural origin and accident conditions. The other is that no reasonable control procedures can achieve significant reductions in individual and collective doses”...

The ARN (acronym for Nuclear Argentinean Regulatory Authority) quantified the grounds of exemption dose criteria in the following terms: individual effective dose in the most exposed individuals should not exceed 10 $\mu\text{Sv}/\text{y}$ and the annual collective effective dose should be below 1man Sv.[2]

The figures mentioned above are based in the fact that 10 $\mu\text{Sv}/\text{y}$ represent an insignificant change in the background radiation and, in the fact that the risk level related to this dose is considered trivial (10^{-7}). The collective dose criteria is founded in the ALARA concept. If the collective dose is small, e.g. on the order of one man Sv per year, protection is often assumed to be optimized, given that regulatory provisions will produce little or no improvement in dose reduction.

Considering that there are some situations where the total activity or the activity concentration involved is trivial enough that the radiological risks incurred from the use, misuse and subsequent disposal are too small to warrant regulatory concern, the Commission of the European Communities developed “Generic Exemption Levels”. If these figures are not exceeded the exemption may be automatic. Therefore, if the regulatory body adopts them, the users don’t have to prove that the practices or sources they manage fulfill the exemption criteria. These values were later adopted by the IAEA in the BSS 115[3] and are applicable to moderate quantities of radioactivity.

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Exemption is only granted to the usage of sources or practices which are previously justified.

Practices likely to be exempted may include the following:

- Surface density gauges (β emitters)
- Testing the integrity of semiconductors and leak testing generally
- In education (eg, sealed sources for demonstrating properties of radiation)
- Technological application (eg, ^{63}N in gas chromatography)
- Smoke detectors (eg, ^{241}Am)
- Research laboratories (eg, C and P as tracers in biochemical research)
- Hospital laboratories (eg, radio-immunoassay techniques)

This list is not exhaustive and some other practices may be relevant for the application of these Generic Exemption Levels.

2. Methodology

The Commission of the European Communities developed different scenarios, described in detail in the document “Radiation Protection 65”(RP-65 [4]), published in 1993, from which Generic Exemption Levels were derived. This document was based on the work undertaken by the National Radiological Protection Board (NRPB), by the Institut de Protection et de Sureté Nucléaire (IPSN) and by the Centre d’Etudes sur l’Evaluation de la Protection dans le Domaine Nucléaire (CEPN). In the RP-65, the Generic Exemption values were calculated for both activity concentration (Bq/g) and total activity (Bq). A total of 3 scenarios and 24 exposure pathways were identified as the most relevant.

A brief description of the scenarios considered is listed below:

- The Normal Use (workplace) scenario represents the use of radionuclides in industry etc, in the manner for which they are intended, and involves external exposure and inadvertent intakes of radioactive materials.
- The Accidental (workplace) scenario represents abnormal procedures or incidents that might occur during the routine use of small amounts of radionuclides. These situations may lead to exposures via a range of external, inhalation and ingestion pathways
- The Disposal (public) scenario represents a member of the public becoming exposed after the disposal of the source. This situation may lead to external, inhalation and ingestion pathways. Both normal and accidental situations are considered.

It should be noticed that the term “*worker*” used in this paper as well as in the RP-65, does not refer to radiological controlled workers, but to individuals that get in touch with radioactive materials as a result of their labor.

3. Generic Exemption Levels And Dose Criteria

Although exemption criteria are based on limits to both individual and collective dose, it should be noticed that the latest does not play a significant role. For example, by limiting the maximum individual dose to 10 $\mu\text{Sv}/\text{year}$ and taking a conservative ratio between the mean and maximum dose of 1/10, the restriction that the collective dose should not exceed 1man-Sievert/year would only be relevant if at least one million individuals were exposed. This situation is almost impossible when dealing with exempted practices, due to the small amounts of radioactive material managed within them.

The Generic Exemption Level for a given radionuclide is defined, as the ratio between the Annual Individual Dose Criteria (Sv/y) and the Dose per unit Activity (Sv/Bq) or the Activity Concentration (Sv/(Bq/g)), as following:

$$\text{Generic Exemp Level (Bq or Bq/g)} = \frac{\text{Annual Individual Dose Criteria (Sv/y)}}{\text{Dose (Sv/y) per unit Activity (Bq) or Act. Concentration (Bq/g)}}$$

The Annual Individual Dose adopted corresponds to an effective dose of 10 µSv. In case of skin irradiation an equivalent dose of 50 mSv/year is considered.

For potential exposures (fire, spillage) an effective dose of 1 mSv with a probability of 10⁻² per year was adopted. Doses to individuals in the workplace and to members of the public are obtained for an activity concentration of 1 Bq/g and a total activity of 1 Bq. It is assumed that the initial inventory of radioactive substances remains at any time constant. Each of the scenarios considered give rise to doses via one or more pathways. Those from the relevant pathways are summed to get the total scenario's dose, which then is used to derive the Generic Exemption value.

The dose per unit activity or activity concentration, is calculated by the following formula, which adjusts through different parameters to each exposure pathway.

$$D = (A \text{ or } C) \times f \times T \times R \times U \times s$$

where:

The term **D** is the equivalent dose for skin doses, the effective dose or committed effective dose for intakes of radionuclides (Sv/y)

The terms **A** and **C** are the activity (1 Bq) or activity concentration (1 Bq/g) respectively.

The term **f** is the fraction of A or C which contributes to the dose, D.

The term **T** is the time for which an individual is exposed to the source (h/y)

The term **R** is the radionuclide dependent dose factor, for a given pathway. This factor may be modified by a geometry factor if the size of the source is smaller than the geometry assumed when deriving the dose factors.

The factor **U** is intended to convert A or C into units consistent with those of the dose factor, R. This conversion depends on the physical properties of the source, e.g. mass, surface area and the form of the source at the time of exposure.

The term **s** represents the probability of an exposure occurring in a year. This is used in situations where it is not probable that a dose will occur in a year, i.e. Accident (workplace) scenario and some Disposal (public) pathways. The probability chosen for all these situations was 10⁻² year.

Annex N°1 shows Generic Exemption Levels for Activity Concentration (Bq/g) and for Total Activity (Bq), for radionuclides of common use.

4. Considerations About The Scenarios

The scenarios considered embrace a wide range of potential exposures that may rise from the inadvertent usage of radioactive material.

Some radionuclides considered have decay products (daughters) which are themselves radioactive and need to be taken into account when assessing exposure. The daughters considered have half-lives sufficiently short, relative to their parents so that secular equilibrium would be likely to be established within the timescales considered in the exposure scenarios.

When evaluating mixtures of artificial origin radionuclides, the following condition has to be accomplished for granting exemption:

$$\sum_{i=1}^n \frac{C_i}{(\text{Activity Concentration})_i} \leq 1$$

Where:

- C_i is the activity concentration (Bq/g) of the *artificial origin* radionuclide “i”.
- **(Activity Concentration)** is the exemption value for the radionuclide “i”.
- **n**: number of radionuclides present in the material.

When evaluating mixtures of natural origin radionuclides, the activity concentration of each radionuclide should be less than the relevant value of the activity concentration given in Table 1.

Table 1: Activity Concentration Values For Natural Radionuclides

Radionuclide	Activity Concentration (Bq/g)
^{40}K	10
Any other natural origin radionuclide	1

The Generic Exemption Levels for ^{238}U and ^{232}Th contemplate all their decay products when assessing the exposure, given the fact that they are naturally found in secular equilibrium.

When concerning mixtures of artificial and naturally occurring radionuclides, both conditions mentioned above should be accomplished in order to be exempted. A similar criterion could be applied for Total Activity exemption.

5. Considerations About The Critical Pathways

For α emitters, usually the critical pathway for exempt activity concentration is inhalation of dust and aerosols in the workplace, and for exempt activity is inhalation of dust or volatiles in the workplace due to an accidental situation.

For pure β emitters, generally, the critical pathways for exempt concentration are accidental ingestion from a member of the public or external exposure in the workplace. For exempted activity, the critical pathway is skin irradiation in the workplace.

For γ emitters the critical pathways in both cases is external exposure of workers.

The main characteristics of the critical pathways are summarized below:

5.1 External exposure from handling a source

Typical situations, which involve handling sources, may include the following:

- Manipulation of small sources, e.g. the fitting of sources into jigs for calibrating instruments.
- Packaging of radioactive sources or materials.
- Machining of small components, e.g. items manufactured from uranium.

The individual is assumed to pick up and handle a source for a limited proportion of the working day (approximately 2-3 minutes). It was also assumed that the source is held by the fingers or within the

palm of the hand, where the skin thickness is 400 μm . For beta radiations the dose rate factor for 400 μm was used and for gamma radiation the dose rate factor adopted was 70 μm . Finally, it was assumed that the glass vial containing liquids, attenuates beta emitters through a glass wall thickness of 150 mg/cm^2 .

5.2 External exposure from a point source

The operator is assumed to be working near a small source, represented by a point source at 1 m. Typical situations where this scenario may occur are as follows:

- When repetitive use is required from a small source to test equipment.
- During fitting of small sealed sources into devices (eg, smoke detectors).
- When small sealed sources or small quantities of unsealed radioactive solutions (vials) may be packaged into containers.
- Use of radioactive sources in industry for tracer studies.

The exposure time adopted for liquids and dispersible solids is 100 h/y and for non-dispersible solids, capsules and foil is 200 h/y.

5.3 Fire: Inhalation of dust or volatiles

This pathway considers the accidental situation in which a laboratory gets on fire and a person inhales dust or volatiles for a period of 10 minutes. This could occur even after the fire is extinguished if the air remains laden with combustion products. It is assumed that the combusted fraction (100% for gases and liquids, 1% for solids) fills a room of 32 m^3 and the air concentration remains constant during the exposure time.

5.4 Fire: external exposure from combustion

In this scenario it is assumed that the fires forms a cloud which persists for at least 10 minutes, in which time an individual immersed in it, will be exposed to an external dose from the gamma and beta emitters. It is assumed that 100% of the combustible fraction fills a room of 32 m^3 and the air concentration remains constant during the exposure time.

5.5 External exposure from a 1 m^3 source

An example of this situation would be the exposure from a small stock piles of ores containing natural radionuclides, process materials or a store of small sources of waste. The operator is assumed to be exposed from a source of 1 m^3 for 100 hours per year.

5.6 Ingestion of an object from a landfill site

In this scenario, a member of the public is assumed to be walking over a landfill site and inadvertently ingests a small quantity of activity from the source. Typical situations considered by this scenario are: a person finding a radioactive source or an object contaminated with radioactivity which has seeped from a source, a person ingesting contaminated soil from their hands or a child accidentally swallowing a contaminated object.

It is assumed that an individual member of the public ingests 1 g of the source per year.

6. Assessing The Adopted Parameters

The parameters used for the dose calculation in the different exposure pathways were analyzed in order to assess their conservativeness. The main observations found are summarized below:

- When calculating the doses due to inhalation, in all cases particles with $1\mu\text{m}$ AMAD have been adopted, this means, particles that reach the lung alveolus. However, only a small fraction of the spectrum of the atmospheric aerosols have the possibility of reaching the lung alveolus, since most of them are retained in the nasals fosses and superior respiratory tract.
- The airborne dust concentrations adopted, for workers as well as for public, are high and some exceed the limit of what it would be considered a tolerable atmosphere.
- For accidental situations, it is assumed that 1.10^{-5} of the total dispersed activity is incorporated. The value usually adopted in safety assessments is 1.10^{-6} . [5-6-7-8]
- The effective dose rate for gamma and beta radiation ($R_5 + R_6$) were calculated by the Monte Carlo Method. In this assessment, the dose reduction due to the site's irregularities where the radionuclides disperse, was not taken into account resulting in overestimated coefficients.[5]
- The exposure times considered in the situations described above are high and appear to be unrealistic. The same happens when estimating the recreation hours of public in landfill sites.
- For the surface of the skin a value of 1 m^2 is adopted, while the correspondent to the reference man is $1,8-2\text{ m}^2$. [9]
- For dose hand calculation, the value of skin thickness adopted for the palm of the hand is ($400\text{ }\mu\text{m}$) and for the face ($40\text{ }\mu\text{m}$), which are values considerably smaller than the correspondent to a reference man ($500-650\text{ }\mu\text{m}$ y $50\text{ }\mu\text{m}$ respectively).[9]

It can be assured that conservative parameters have been adopted, maximizing the resultant doses. In addition, it should be stressed out, that the physico-chemicals forms adopted in every case were the most conservative ones, which also imply an overestimation of the dose.

7. Conclusion

- From the analysis of the scenarios, it may be concluded that the physico-chemical forms, the geometric and occupational factors, and the scenario's specific parameters are based in a conservative criteria overestimating the effective doses.
- The scenarios considered contemplate a wide range of potential exposures that may rise from the inadvertent usage of radioactive material.
- These values are not expected to be modified in the next revision of the BSS 115 given the fact that they arise from the individual dose criteria.
- According to the assessment carried out, it was suggested that the ARN should adopt the use of Generic Exemption Values, in order to improve the regulatory management's efficiency and to optimize the utilization of its human and economic resources. This initiative is presently in course of implementation.

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Annex N° 1

Generic Exemption Levels For Some Radionuclides

These values were taken from the document “Radiation Protection 65”

Radionuclide	Generic Exemption Level per unit of Activity Concentration (Bq/g)	Critical Pathway for Exempt Activity Concentration	Generic Exemption Level per unit of Activity (Bq)	Critical Pathway for Exempt Activity
H-3	1,00E+06	Ing Acc (P)	1,00E+09	Ing Acc (P)
C-14	1,00E+04	Ing Acc (P)	1,00E+07	Ing Acc (P)
F-18	1,00E+01	Ext (W)	1,00E+06	Ext (W)
P-32	1,00E+03	Ext (W)	1,00E+05	Skin (W)
S-35	1,00E+05	Ing Acc (P)	1,00E+08	Ing Acc (P)
Cr-51	1,00E+03	Ext (W)	1,00E+07	Skin (W)
Ni-63	1,00E+05	Ing Acc (P)	1,00E+08	Ing Acc (P)
Se-75	1,00E+02	Ext (W)	1,00E+06	Ext (W)
Kr-85	1,00E+05	Extg (W)	1,00E+04	Skin (W)
Sr-90	1,00E+02	Ext (W)	1,00E+04	Skin (W)
Mo-99	1,00E+02	Ext (W)	1,00E+06	Ext (W)
I-125	1,00E+03	Ext (W)	1,00E+06	Ing Acc (P)
I-131	1,00E+02	Ext (W)	1,00E+06	Ing Acc (P)
Cs-137	1,00E+01	Ext (W)	1,00E+04	Skin (W)
Pm-147	1,00E+04	Inh (W)	1,00E+07	Ing Acc (P)
Ir-192	1,00E+01	Ext (W)	1,00E+04	Skin (W)
Au-198	1,00E+02	Ext (W)	1,00E+06	Ext (W)
U-234	1,00E+01	Inh (W)	1,00E+04	Inh Acc (P)
U-238	1,00E+01	Inh (W)	1,00E+04	Inh Acc (P)
U-238N(**)	1,00E+00	Inh (W)	1,00E+03	Skin (W)

References

Activity Concentration

- **Ing Acc (P)** = Accidental Ingestion for the Public (landfill)
- **Ext (W)** = External Radiation by a 1m³ contaminated source (workplace)
- **Extg (W)** = External Radiation by a bottle (workplace)
- **Inh (W)** = Inhalation (workplace)

Activity

- **Ext Acc F (W)** = External Radiation in workplace due to an accidental fire
- **Skin (W)** = Skin Doses in workplace
- **Inh Acc F (W)** = Inhalation in workplace
- **Ext (W)** = External Radiation in workplace (skin effective dose + point source)
- **Inh Acc (P)** = Accidental Inhalation from the Public (landfill)
- **Ing Acc (P)** = Accidental Ingestion from the Public (landfill)

(**) In case of minerals containing natural radionuclides.

Norm Survey in Argentina

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Norm survey in Argentina

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Abstract. The oil and gas industry, which is especially significant in Argentina, is one industry that concentrates natural radionuclides during its processes. In addition, there are a few underground mines under development, where radon levels may be high. The Nuclear Regulatory Authority of Argentina (ARN) carried out a project with the objective of evaluating NORM, mainly in these types of industries. Eight facilities were characterized, three related to the gas industry, four related to the oil industry and a survey of radon gas in a gold underground mine. First, background measurements were made and then a screening survey was carried out to detect values above background. Of the values obtained, 57% were in the background range, 19% were below 2 $\mu\text{Sv/h}$, 15% were in the range 2–10 $\mu\text{Sv/h}$ and 9% were above 10 $\mu\text{Sv/h}$. Some values were as high as 400 $\mu\text{Sv/h}$. The annual effective doses were estimated to be in the range 0.02–1.6 mSv/a. Samples were taken and later analysed by gamma spectrometry, liquid scintillation and fluorimetry. It was confirmed that the main radionuclides involved in the oil and gas extraction process are ^{226}Ra and ^{228}Ra . The radium isotope concentrations measured in some samples were above the exemption values established by the International Basic Safety Standards. Elevated radon levels were detected in gas facilities and in the gold mine under development. The values obtained in gas facilities showed that radon concentrates in the ethane and propane flows. As the flows in the gas industry are confined, it does not mean an exposure during normal operation. In the case of the gold mine, the values detected were informed to the pertinent authorities as well as the facility in order to take actions to reduce concentrations below the action levels. Finally, protective measures to reduce occupational doses in the cleaning and maintenance processes were suggested, as well as for storage of NORM-contaminated items.

KEYWORDS: *norm, natural radionuclides, oil and gas industry, underground mine, radon*

1. Introduction

Radioactive materials containing radionuclides of natural origin are known as NORM (naturally occurring radioactive material). Some minerals have significant levels of natural radionuclides that are extracted and processed with other elements. Some industries involve processes that concentrate natural radionuclides and then may cause some risk to people if the exposures are not under control.

NORM are found in some effluent flows and wastes from some non-nuclear industries, for example in metal residues, scales, sludges and fluids. These materials, the by-products and the final products from processes may enhance the exposure of workers and members of the public. The most important radioactivity source in NORM is due to the presence of isotope products of the uranium and thorium decay chains [1–3].

The presence of radioactive materials of natural origin in geologic formations is well known. The materials containing natural radionuclides found in oilfields are typically located in subsurface formations of oil and gas reservoirs created in the Jurassic period. In the oil and gas industry the techniques used in forcing the oil to the surface include recirculation of produced water, which is extracted with the final products. The NORM materials are transported to the surface with this produced water. A decrease in pressure and temperature results in sulphate and carbonate precipitation inside the pipelines and in the internal surfaces of the equipment. The similar chemical behaviour of radium and barium produces selective co-precipitation of both elements in scales. Other products of the uranium and thorium decay chains can also be found. The naturally radioactive material which is not present in scales appears in the vessels with the drained water or in sludges. Other radionuclides of interest, particularly in gas equipment, are radon gas and ^{210}Pb , which usually forms a thin cap in the

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internal surface of processing equipment [4, 5]. In the case of underground mines radon gas may concentrate up to high levels, particularly in the exploration stage.

From the occupational point of view, the main aspects of radiological protection related with scales and sludges are gamma irradiation and internal contamination of workers arising in the maintenance of equipments containing NORM. In relation with underground mines the exposure of workers by inhalation of radon gas may be significant.

The Nuclear Regulatory Authority of Argentina (ARN) carried out a project whose objective was the evaluation of NORM, mainly in these types of industries. With this purpose, eight companies were characterized; three of them related with gas industry, four with oil industry and one underground gold mine.

In this work the results obtained within the companies surveyed are presented with the aim of evaluating the presence of NORM and the exposure of workers.

2. Facilities description

2.1. Oil facilities

2.1.1. Facility A

The company provides pumping systems for oil and gas extraction processes. This facility performs the assembling of equipments with new or recovered pieces. The equipments to recycling arrive to a sector called “discharging” and from there go to the “disassembling” sector, where the components are washed, recovered and refurbished. The rejected components return to the discharging sector to await disposal as waste or selling as scrap.

2.1.2. Facilities B, C and D

These facilities perform services of washing, maintenance and inspection of tubing. They are different bases of the same company. In our country, the company has seven bases.

The tubes arrive and are classified and stored in the store area until the washing process begins in the washing area. The wastes from the washing process are temporarily stored until they are removed by the service companies. All processes are performed in well-ventilated areas. The washing process is carried out in two steps: first the tubes are introduced in a washing container with a mix of water and gasoline at 90°C, during 10 to 15 minutes. Then an internal and external manual washing with pressured water is carried out. The remaining water is collected in vessels called API. In some facilities it is also used a mechanic equipment to remove scales. The solid wastes from the process are collected in two containers located at both ends of the pipe. These wastes are then transferred to a large container where they are temporarily stored.

2.2. Gas facilities

2.2.1. Facility E

The company separates and fractionates the heavy components of natural gas (LGN) in two facilities: a separation plant and a fractionation plant. In the separation plant the natural gas is received, dried and liquified. Then the liquids are sent by a special pipeline to the fractioning plant, where ethane, propane, butane and gasoline are separated. In the fractioning plant there are five main areas: reception of the rich components mix, separation of the rich components, ethane reconditioning, storage areas, dispatch and services.

The distillation process is performed in three continuous stages:

- (a) A de-ethanizing tower retains ethane at the top.
- (b) A de-propanizing tower retains propane at the top.
- (c) A third tower retains butane at the top and gasoline at the bottom.

Then, ethane is purified and dispatched, while propane, butane and gasoline are transitory stored.

2.2.2. Facilities F and G

These two facilities produce ethylene and polyethylene. The ethylene is obtained from ethane. The polyethylene is produced from ethylene. Facility F has been in operation since 1981 and facility G since 2001.

2.3 Underground mine

This gold facility is an underground mine. At the moment radon gas measurements were performed, the facility was evaluating the ventilation system efficiency.

3. Measurements

In the case of oil and gas facilities, to determine whether there were areas or equipments contaminated with NORM, various locations were surveyed. The sampling points were selected on the basis of the processes performed in each place, taking into account the origin, function and visual inspection of the different items. In situ dose measurements were performed and samples were taken for analysis at the ARN laboratories. In gas facilities radon gas was also measured in the different separation streams. In the case of the underground mine radon gas measurements in air were carried out at different locations inside the mine, including all the galleries.

3.1. In situ measurements

Dose rate measurements were carried out in predetermined areas in agreement with the processes performed in each facility. The equipment used was:

- (a) Scintillation detector NaI(Tl) IDENTIFINDER 1,2''x 1,5''
- (b) Geiger- Müller detector AUTOMESS 2174

First of all background measurements were performed in the surroundings of each facility. Then, in facilities A, E, F and G measurements were performed in contact, with the locations being selected on the basis of the origin, function (information given by the facility staff) and visual inspection of the elements (sludge presence). If possible, the pieces were also evaluated in its internal surface (with a probe). The different measurements are summarized in Table 1.

Table 1: Dose rate measurements in contact at facilities A, E, F and G.

Facility	Background measurement ($\mu\text{Sv}\cdot\text{h}^{-1}$)	Range of dose rate values ($\mu\text{Sv}\cdot\text{h}^{-1}$)	Number of measurements in range
A	0.20 ± 0.02	Background level	9
		< 2 $\mu\text{Sv}\cdot\text{h}^{-1}$	5
		2 - 10 $\mu\text{Sv}\cdot\text{h}^{-1}$	9
		10 - 20 $\mu\text{Sv}\cdot\text{h}^{-1}$	2
		> 20 (28.2 and 30 $\mu\text{Sv}\cdot\text{h}^{-1}$)	2
E	0.10 ± 0.02	Background	7
		< 1 $\mu\text{Sv}\cdot\text{h}^{-1}$	6
F	0.15 ± 0.04	Background	9
		< 2 $\mu\text{Sv}\cdot\text{h}^{-1}$	11

		2 - 10 $\mu\text{Sv}\cdot\text{h}^{-1}$	5
		> 10 $\mu\text{Sv}\cdot\text{h}^{-1}$ ^a	5
G	0.12 ± 0.03	Background	19
		< 1 $\mu\text{Sv}\cdot\text{h}^{-1}$	11
		1 - 3 $\mu\text{Sv}\cdot\text{h}^{-1}$	16

^a See table 2 for details

In Table 2 values above 10 $\mu\text{Sv}\cdot\text{h}^{-1}$ found in F facility are presented separately

Table 2: Dose rates exceeding 10 $\mu\text{Sv}\cdot\text{h}^{-1}$ at F facility.

Sampling points	Dose rate in contact ($\mu\text{Sv}\cdot\text{h}^{-1}$)	Dose rate at 1 meter ($\mu\text{Sv}\cdot\text{h}^{-1}$)	Dose rate at 3 meters ($\mu\text{Sv}\cdot\text{h}^{-1}$)
P5601 pump	400	20.0	2.0
P5601 suction pump	320	20.0	-
Pipes at 1 meter from P5601 pump	110	-	-
Pipes at 2 meters from P5601 pump	30	-	-
5601 pipe	22	5.5	-

In B, C and D facilities dose rate screening was performed in the surroundings of each area. This screening was performed with the objective of detecting dose rate values above background. After that, detailed measurements were performed at those points where values above background were found. The results are summarized in Table 3.

Table 3: Dose rate measurements in contact at facilities E, F and G.

Facility	Background measurement ($\mu\text{Sv}\cdot\text{h}^{-1}$)	Points above background	Dose rate values in contact ($\mu\text{Sv}\cdot\text{h}^{-1}$)
B	0.09 ± 0.01	1	2.2
C	0.11 ± 0.01	0	-
D Store area		1	2.8
D Washing area ^a	0.13 ± 0.01	3	1-10
		1	10-20

^a See table 4 for details

Table 4 specifies the values found in the washing area of D facility.

Table 4: Measurements in the washing area at facility D

Sampling points	Dose rate values in contact ($\mu\text{Sv}\cdot\text{h}^{-1}$)	Dose rate values at 1 meter ($\mu\text{Sv}\cdot\text{h}^{-1}$)	Dose rate values at 3 meters ($\mu\text{Sv}\cdot\text{h}^{-1}$)
Washing container	1.0	-	-
Large container	10.0 – 18.5	3.0	0.900
Waste container 1	1.0 – 2.8	-	-
Waste container 2	3.8	0.800	-

3.2. ARN laboratories measurements

Samples from scales, sludges and washing effluents from facilities A, B, C and D were analyzed in the ARN laboratories later. The scales and sludges samples were obtained from pieces whose dose rate measurements resulted above background.

First, the samples were analyzed by gamma spectrometry using Canberra GeHp detectors, model GX2518 with 30 % efficiency. Then, Ra-226 analyses were performed by a radiochemical method, based on the co-precipitation of radium with BaSO₄ and the measurement of radon gas by liquid scintillation. Uranium concentration was measured by fluorimetry using a Jarrel Ash equipment. The results are summarized in Table 5.

Table 5: Maximum and minimum radium isotopes and natural uranium concentration values in samples from facilities A, B, C and D

Facility	Uranium		²²⁶ Ra		²²⁸ Ra	
	Minimum value	Maximum value	Minimum value (Bq·g ⁻¹)	Maximum value (Bq·g ⁻¹)	Minimum value (Bq·g ⁻¹)	Maximum value (Bq·g ⁻¹)
A	< 0.4 µg·g ⁻¹	1.9 ± 0.8 µg·g ⁻¹	< 0.1	1270 ± 130	115 ± 11	1670 ± 17
B	< 10.0 µg·l ⁻¹	33.0 ± 9.8 µg·l ⁻¹	< 1.7 E-3	26.8 ± 2.7	< 1.1 E-3	9.6 ± 0.9
C	< 10.0 µg·l ⁻¹	1.5 ± 0.7 µg·g ⁻¹	< 1.4 E-3	0.07 ± 0.01	< 9.6 E-4	0.1 ± 0.01
D	< 0.4 µg·g ⁻¹	< 0.7 µg·g ⁻¹	1.9E-3 ± 4E-4	18.7 ± 1.8	2.1E-3 ± 4E-4	65.4 ± 6.5

In E, F and G facilities, radon gas measurements were performed by Lucas cell method. This method consists in collecting air samples in cells coated with SZn(Ag) and then the cells are measured using Ludlum 2200 alpha counters. The results are presented in Table 6.

Table 6: Radon gas concentrations in the different gas streams at facilities E, F and G

Facility	Radon gas concentration (Bq·m ⁻³)	Sampling points
E	1841 ± 300	Ethane + CO ₂
F	337773 ± 30000	Tower top (propane 18% - propylene 75%)
G	62572 ± 5000	Tower top (propane 18% - propylene 75%)

In the underground mine surveyed, radon gas in air was measured at different locations inside the mine, including all the galleries (points 1 to 10). The measurements were performed using vials containing activated charcoal. Radon gas was adsorbed on the charcoal and, after that, a scintillation cocktail was added. Finally the vials were measured by liquid scintillation. The results are presented in Table 7.

Table 7: Radon gas concentrations in air of an underground mine

Sampling points	Radon gas concentration (Bq·m ⁻³)
1	1840
2	3460
3	8200
4	1280
5	180
6	8200

7	6240
8	12900
9	145
10	150

4. Results

4.1. External exposure

Dose rate values above background were detected in tubing containing scales, in isolated pieces, in containers with material from washing and maintenance processes and in ethane and propane flows. It was found that 57 % of these dose rates were at background levels, 19 % were below $2 \mu\text{Sv}\cdot\text{h}^{-1}$, 15% were in the range $2\text{--}10 \mu\text{Sv}\cdot\text{h}^{-1}$ and 9% were above $10 \mu\text{Sv}\cdot\text{h}^{-1}$.

In order to assess the maximum occupational dose that a worker might receive in these facilities, conservative scenarios were defined. Occupancies were calculated on the basis of information given by the facilities staff. Homogeneous whole body irradiation was assumed. The maximum dose rate measurements, occupancies and annual effective doses calculated in each case are shown in Table 8:

Table 8: Results of external exposure assessments

Facility	Pieces above background	Maximum dose rate ($\mu\text{Sv}\cdot\text{h}^{-1}$)	Occupational factor (hours \cdot y ⁻¹)	Annual effective dose (mSv \cdot y ⁻¹)
A	Isolate pieces, pipes	30	20 (5 minutes per day - 240 days in a year)	0.6
B	Pipes	2.2	25 (5 minutes per day - 300 days in a year)	0.05
C	None	-	-	-
D ^a	Pipes	2.8	25 (5 minutes per day - 300 days in a year)	0.07
	Container with sludges	0.8 ^b 18.5 ^c	320 25 (5 minutes per day - 300 days in a year) in contact	0.26 0.45
	Large container	3 ^b	50 (10 minutes per day - 300 days in a year) at 1 meter	0.15
E	Depropanizer pump	0.9	20 (5 minutes per day - 240 days in a year)	0.02
F	Pump 5601	400	4	1.6
G	Pump P93	3.0	4	0.01

^a In the case of D facility it is assumed that a worker may be exposed to all the scenarios, being the total annual effective dose $0.93 \text{ mSv} \cdot \text{y}^{-1}$

^b Dose rate at 1 meter.

^c Dose rate in contact.

4.2 Internal contamination

4.2.1 Oil and gas facilities

The incorporation of radioactive material is an exposure pathway that becomes important during washing and maintenance processes, in which workers may intake by inhalation particulate material. It was not possible to evaluate these pathways, as the facilities were not performing maintenance duties during the investigations.

It was confirmed from the measurements carried out in the gas facilities that radon gas is concentrated in ethane and propane streams. This is a result of radon having a condensation point between those of propane and ethane and thus follows these products in distillation and cracking flows.

4.2.2 *Underground mine*

From the radon gas measurements performed inside the mine it can be seen that, in the sampling points named 1, 2, 3, 4, 6, 7 and 8, the values resulted above the corresponding action level values established for workplaces ($1000 \text{ Bq}\cdot\text{m}^{-3}$) [6-8].

4.3. *Sample analyses in ARN laboratories*

The analyses performed by fluorimetry showed that uranium is not concentrated in the sludges. This reflects the fact that uranium is almost not mobilized in the oil extraction process.

The analyses performed by gamma spectrometry confirmed that the radionuclides involved come from the decay chains of U-238 and Th-232. The radionuclides that mainly concentrate in these processes are Ra-226 and Ra-228. Some of the radium isotopes concentrations measured were above the exemption values established by BSS 115 [6], namely $10 \text{ Bq}\cdot\text{g}^{-1}$ for Ra-226 and Ra-228).

5. Conclusions

The dose rates measured at most facility locations were within normal background levels. Some of the points, which resulted above background, were from tubing with NORM (facilities A, B, D) and from the washing area in facility D. The wastes arising from the washing area are stored in each facility until removal by services companies. It was reported that this material might be used in road constructions. In gas facilities (E, F and G) some dose rates resulted above background in the ethane and propane flows.

In oil facilities, an annual effective dose of $0,6 \text{ mSv}\cdot\text{y}^{-1}$ was conservatively estimated from the highest dose rate measured in tubing. In facility D, assuming that a worker may be exposed to additional scenarios, including duties, not only in the store area but also in the washing area, the annual effective dose calculated in a conservative way was $0.93 \text{ mSv}\cdot\text{y}^{-1}$. It is suggested that the doses received by the workers in these areas be optimized by examining the possibilities for reducing the occupancy times.

In relation with gas facilities, the values measured in facility F were higher than those measured in facility G, owing to a greater surface accumulation of radionuclides in the piping of the older facility. The annual effective dose calculated in a conservative way from the highest value measured was $1.6 \text{ mSv}\cdot\text{y}^{-1}$. Although the time spent by workers in the areas of highest dose is short, it was suggested that the presence of workers in these areas be justified and their doses be optimized by examining the possibilities for reducing the occupancy times.

It is important to point out that the results obtained in this investigation may not agree with the results of future surveys, due to the fact that the contamination of tubing and different pieces may vary over time. All the annual effective doses calculated (external exposure) are very low in comparison with the dose limit established in the ARN Standards for workers ($20 \text{ mSv}\cdot\text{y}^{-1}$). In the case of facility F, the value calculated exceeded the limit for members of the public ($1 \text{ mSv}\cdot\text{y}^{-1}$) [6]. In order to have a

better dose assessment for workers it would be advisable to perform Thermoluminescence detection (TLD) measurements during three months and to evaluate the inhalation and ingestion pathways, specially during inspection, repair or maintenance activities, because aerosols may be generated in these processes.

For that pieces with dose rate measurements above background it would be important to define the suitable storage methods. As some pieces are sold as scrap to other people it is advisable to perform a previous washing and evaluation to reduce the dose rate level. In this sense, some washing and maintenance procedures were suggested to the facilities based on international bibliography [9-10].

From the measurements performed in the laboratories, it was confirmed that the radionuclides found in these type of industry were Ra-226 y Ra-228, members of the U-238 and Th-232 decay chains. In some cases, the radium isotopes measured were above the exemption values established in BSS 115. This work has also confirmed that uranium is not mobilized in the oil extraction process. On the basis of radon gas measurements performed in gas facilities it was confirmed that radon concentrates in ethane and propane flows. The possibility of gas inhalation should be taken into account during inspection, repair or maintenance activities, as in normal operation the gas is confined in the pipes and vessels with no risk to workers.

In relation with the underground mine, the action level for remedial action relating to chronic exposures situations involving radon in workplaces is $1000 \text{ Bq}\cdot\text{m}^{-3}$ [6-8], assuming an equilibrium factor (F) between the gas and its progeny of 0,4. Therefore, in that locations where radon measurements result above this action level it has to be adopted some actions so as to reduce radon concentration values below the action level.

In several sampling points of the mine radon gas concentration resulted above the action level. Therefore, it was recommended to the facility to improve all the ventilation system, especially in the mentioned locations. Once this stage is fulfilled, in order to check the ventilation effectiveness it was informed to the staff that a new radon gas survey should have to be carried out.

Finally, for the industries analyzed, it was suggested that the facilities be re-evaluated to determine the buildup of Norm contamination over time.

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Working towards Residential Radon Survey in South America

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Working towards Residential Radon Survey in South America

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Abstract. Information about residential radon levels in low and middle income countries is very sparse. In response to the World Health Organization initiative in the International Radon Project, we propose a research project that will address this knowledge gap in South America by conducting a residential radon survey. Following initial in vitro and in vivo studies of radon and studies of uranium miners exposed to radon, over twenty large case-control studies of lung cancer risk from exposure to residential radon have been completed worldwide by year 2004. Recently pooled data from these individual studies have been analyzed. These collaborative analyses of the indoor studies in Europe, North America, and China provide strong direct evidence that radon is causing a substantial number of lung cancers in the general population. To reduce radon lung cancer risk, national authorities must have methods and tools based on solid scientific evidence to develop sound public health policies. We propose to conduct a survey in ten South American countries using the distribution and analysis of passive alpha tracking detectors in houses selected at random in pre-selected cities in each participating country. We also present an approach to estimate the cost of carrying out such a survey and the radon laboratory infrastructure needed. The results of the proposed survey will allow to conduct assessment of the exposure to residential radon in the populations of South American countries and to assess the health impact of this exposure. The results of the project will also help national health authorities in developing national residential radon action levels and regulations, as well as provide public health guidance for radon awareness and mitigation.

KEYWORDS: *radon, residential radon, risk, lung cancer, survey, South America, policy*

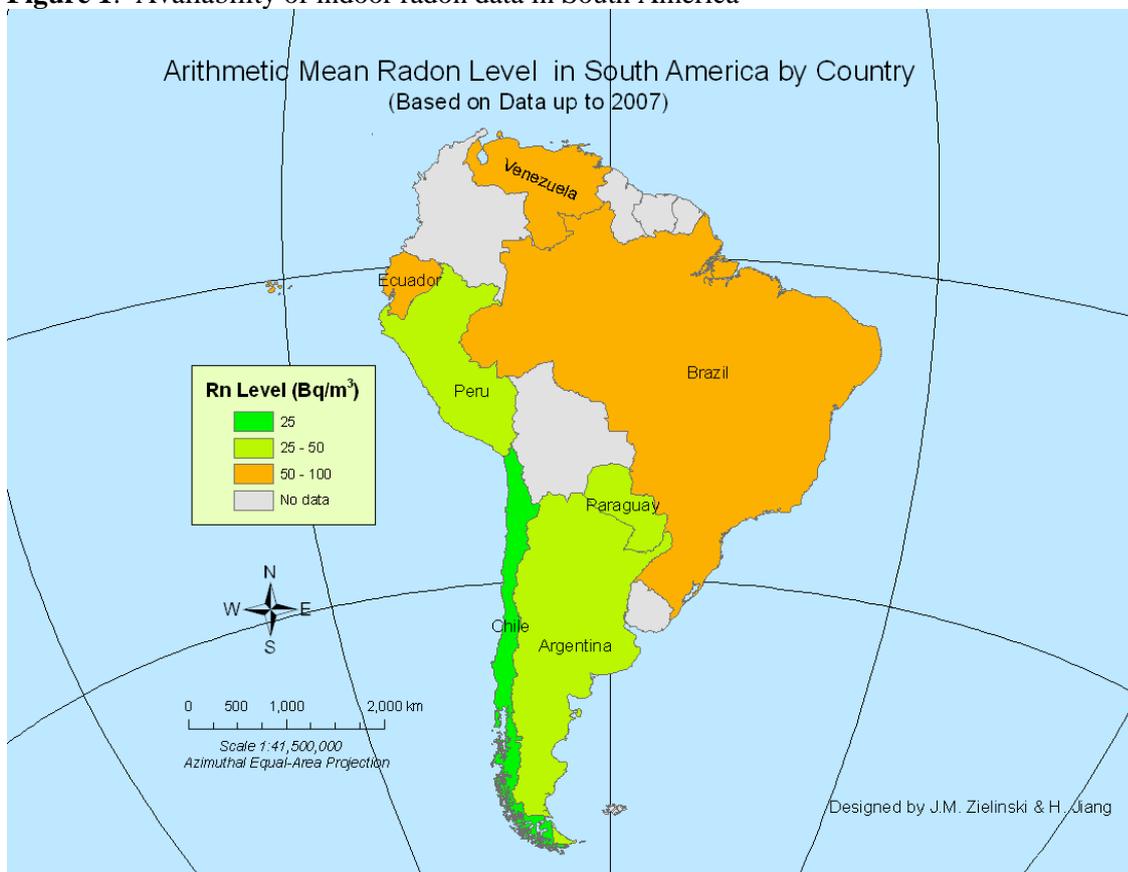
1. Background

Radon is a chemically inert, naturally occurring radioactive gas without odor, color or taste. It is produced from radium in the decay chain of uranium, an element found in varying amounts in all rocks and soil all over the world. Radon gas escapes easily from the ground into the air and disintegrates through short-lived decay products called radon daughters or radon progeny. The short-lived progeny, which decay emitting heavily ionizing radiation called alpha particles, can be electrically charged and attach to aerosols, dust and other particles in the air we breathe. As a result, radon progeny may be deposited on the cells lining the airways where the alpha particles can damage DNA and potentially cause lung cancer [1]. An increased risk of lung cancer is the main health hazard from high radon exposure. This has been substantiated in many studies of uranium miners [1]. Based on these studies, the International Agency for Research on Cancer (IARC), a WHO agency

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specializing in cancer, and the US National Toxicology Program have classified radon as a human carcinogen [2-4]. The health hazard of much lower levels of radon found in homes and other places has also been investigated. By the year 2004, over twenty large case-control studies of lung cancer risk from exposure to residential radon have been completed around the world [5-7]. Recently researchers have pooled the information from these individual studies and re-analyzed the data. These collaborative analyses of the indoor radon studies in Europe [8-9], North America [10-11], and China [12] provide strong direct evidence that radon is indeed causing a substantial number of lung cancers in the general population. To reduce lung cancer risk, national authorities need methods and tools based on solid scientific evidence. In particular, strategies for reduction of radon exposures must be based on information about the geographic variation of indoor radon concentrations within a country, as well as the number and location of homes with high radon levels. The concentration of radon in a home depends on the amount of radon-producing uranium in the underlying rocks and soils, the routes available for its passage into the home and the rate of exchange between indoor and outdoor air. Geographical variations of indoor radon concentrations are related to regional differences in soil composition, climate and other factors [1, 13]. Nationwide surveys of indoor radon concentrations have been conducted mostly in high income industrialized countries. In developing countries, the data about residential radon levels are very sparse.

Figure 1: Availability of indoor radon data in South America



In North America, comprehensive data from nationwide surveys of indoor radon concentrations exist in the United States [14] and Canada [15-18]. In Canada, the national radon survey involved measurements in 14,000 houses in 19 cities (approximately 0.5% of all dwellings in the target cities) [15-16, 18]. This sample is representative of the country with a territory of $\approx 9,000,000$ squared kilometers and with a population of $\approx 33,000,000$ inhabitants.

Measurements of residential radon concentration have been conducted in six South American countries: Brazil [19-23], Argentina [19,24], Ecuador, Peru, Venezuela [19], and Chile [25] (Figure 1). Though the measurements provide useful information on indoor radon levels for some population

groups in selected areas, they are not representative of the entire populations and territories of the respective countries. The largest number of measurements (2034 measurements in 14 cities) was conducted in Argentina [19], a country with a surface area of $\approx 2,780,400$ square kilometers and a population of $\approx 36,000,000$ inhabitants. In Brazil (surface area $\approx 8,547,400$ square kilometers, population $\approx 183,000,000$), radon measurements were conducted in 869 dwellings [23]. These numbers are even smaller in Ecuador (61 measurements), Peru (168 measurements), and Venezuela (143 measurements) [19]. Available information on radon gas concentrations for Chile is limited to measurement in 119 houses in the city of Santiago (population ≈ 5 million), and in several of the 15 houses in the Sub-Antarctic Presidente E. Frei station [25]. To our knowledge, there are no published data on indoor radon concentrations for other South American countries. We propose a research project that will address this knowledge gap by conducting a residential radon survey in South America. The project responds to the World Health Organization initiative on residential radon [26].

2. WHO International Radon Project

In January 2005, the WHO launched the International Radon Project in which over 20 countries have formed a network of partners to identify and promote programs that reduce the health impact of radon [27]. The first meeting of the Project was held in Geneva in January 2005 to develop a strategy for dealing with this important health issue. The key objectives of the Project are to:

- identify effective strategies for reducing the health impact of radon;
- promote sound policy options, prevention and mitigation programs to national authorities;
- raise public and political awareness about the consequences of exposure to radon;
- raise the awareness of financial institutions supplying home mortgages to the potential impact of elevated radon levels on property values;
- monitor and periodically review mitigation measures to ensure their effectiveness;
- estimate the global health impact of exposure to residential radon and so allow resources to be allocated effectively to mitigate the health impact of radon; and
- create a global database (including maps) of residential radon exposure.

3. Methodology of South American Radon Survey

3.1 Scope

Table 1: Population and number of dwellings in each country from South America.

Country	Population	Surface area (km ²)	Number of people / km ²	Year	Number of dwellings
Argentina	36,270,130	2,780,400	13	2001	10,073,625
Bolivia	8,328,700	1,098,581	8	2000	2,082,175
Brasil	183,162,261	8,547,404	21	2005	45,790,565
Chile	15,498,930	756,096	20	2002	3,874,733
Colombia	41,589,018	1,141,748	36	1999	10,397,255
Ecuador	12,645,095	272,045	46	2000	3,161,274
Paraguay	5,734,000	406,860	14	1999	1,433,500
Peru	26,090,000	1,285,216	20	2001	6,522,500
Uruguay	3,360,868	175,016	19	2002	840,217
Venezuela	23,706,000	912,050	26	1998	5,926,500
South America	356,385,002	17,375,416	21		90,102,344

Passive alpha tracking detector will be distributed in houses selected at random in pre-selected cities in each country. Some important issues have to be defined, such as: the number of dwellings in each country, the measurement period, and the logistic to distribute and collect detectors in each country and within countries. Some considerations regarding those issues had to be done in order to estimate the involved costs. Table 1 presents data from each country regarding population size, surface area, population density and an estimation of number of dwellings by considering 4 people in each house.

In Table 2 we summarize administrative features of each country, and provide estimates of the number of dwellings in cities with more than 100,000 inhabitants. We propose to select 0.5% of all dwelling in the target 413 cities with more than 100,000 inhabitants (i.e. 217,000 dwellings) for long term (12 months) radon monitoring. This is a similar sampling rate as in the residential radon survey in Canada that was carried out in 1977, 1978 and 1980 in nineteen Canadian cities [15-16, 18]. The proposed sampling frame should allow the estimation of not only national population averages for participating countries, but also reliable averages for smaller geographical units (Federal States, Departments or Provinces). One of the primary objectives of the proposed survey will be to estimate with a good precision the proportion of the population of households (or individuals) subject to a radon concentration level higher than 200 Bq/m³.

Table 2: Demographics of South America countries. Estimates of number of dwellings in the survey.

Country	Number of Federal States, Departments or Provinces	Number of Main cities ^a	Number of cities > 100,000 inhabitants	Population in cities > 100,000 inhabitants	Number of dwellings in cities > 100,000 inhabitants	0.5% of dwellings cities > 100,000 inhabitants
Argentina	24	21	58	20,780,089	5,195,022	25,975
Bolivia	10	14	9	3,853,490	963,373	4,817
Brasil	27	33	219	87,084,400	21,771,100	108,856
Chile	13	24	24	9,407,530	2,351,883	11,759
Colombia	33	20	38	22,087,318	5,521,830	27,609
Ecuador	22	17	14	5,335,807	1,333,952	6,670
Paraguay	18	14	7	1,498,712	374,678	1,873
Peru	25	11	7	10,329,369	2,582,342	12,912
Uruguay	19	14	1	1,303,182	325,796	1,629
Venezuela	25	22	36	11,955,000	2,988,750	14,944
South America	216	190	413	173,634,897	43,408,724	217,044

^a Some important cities (such as provincial capitals) with less than 100, 000 inhabitants are included in this category

3.2 Radon laboratory infrastructure (capacity building)

The Radon laboratory at the Institute of Radioprotection and Dosimetry (IRD) in Brazil, which uses the tracking etching technique, has a limited capacity of analysis. In order to carry on all analyses for the South America survey, the IRD Radon Laboratory will need to be upgraded with an automatic radon system RADOSYS, which has an analytical capacity of 450 samples per day [28].

Table 3: Institutions in South America countries contacted to take part in a radon survey project.

Country	Contact
Argentina	Lic. Analia Canoba Gerencia Apoyo Cientifico Autoridad Regulatoria Nuclear - Centro Atomico Ezeiza Tel: 54-11-6-779-8363 Fax: 54-11-6-779-8460 E-mail: acanoba@cae.arn.gov.ar
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3.3 Distribution of radon detectors

Recruitment of participants will be the most difficult task. Usually, in most South American countries, owners do not allow strangers to enter their homes (mainly in big cities). Therefore, it would be very difficult to select houses at random and get permission from the owners to install radon detectors. A strategy will need to be developed to overcome this problem. One possibility is to select dwellings through some national or provincial organism, like Army, Federal and State Institutions or some big industries/companies in each city and ask for permission to use employees' residence to do the radon survey. Usually employees' residences are spread over all parts of the city. This strategy does not represent a statistically random collection but could be a good alternative to overcome the difficult task of recruiting participants. Independent of the sampling strategy, we propose to setup a survey team in each city consisting of a project leader and five assistants.

3.4 Partners in South American countries

Table 3 presents a list of Institutions and respective researchers that were contacted by IRD-Brazil to ask about their interest in taking part in a radon survey in South America and also to ask for information about their country. Most were very interested in taking part in the project.

3.5 Evaluation

Evaluation framework for this project will be developed with the office of the director of Pan-American Health Organization.

4. Cost of the program

4.1. Management

It is estimated that the project duration will be four years. All management costs for the project will be covered by in-kind contributions from team members, local collaborators, but this project does not mean an economic funding by the institutions of each country. Therefore international funding will be sought in accordance with the scale of the project.

4.2 Radon Laboratory Infrastructure

In order to carry on all analyses for the South America survey, the IRD Radon Laboratory will need to be upgraded with an automatic radon system an estimated total cost of US\$ 23,736.00.

The number of detectors to be bought will depend on the number of dwellings to be sampled. It was considered that two detectors would be used at the same time per house (one in a living room and one in a bedroom). Therefore, 434,088 detectors will be needed. As the cost of 250 detectors is US\$1206.00, the total cost for 434,088 detectors will be US\$ 2,094,037.

4.3 Transportation

Transportation of the detectors from and to the IRD Radon Laboratory in Brazil was estimated using the cost of a 30 kg international SEDEX. The estimated cost is US\$ 1,500.00 per country. Considering 9 countries, the total cost of transportation would be US\$ 13,500.00. Transportation of detectors from the coordinator Institution in each country to all participant cities in the country was estimated considering the cost of a national SEDEX in Brazil of US\$ 40.00. Considering 413 cities, two-way, total costs would be US\$33,040.00. Total cost of detectors transport would be US\$ 46,540.00.

4.4 Logistic to distribute the detector in each country

The survey team in each city will be comprised of a project leader and five assistants. An estimation was done considering that the salary of each assistant is approximately US\$ 300 per month, 3 months work to complete the detectors installations in each city and 3 months work for collecting the detectors. Five assistants for 6 months would cost US\$ 9000,00 approximately for each city. Therefore, logistic of distributing and collecting detectors in 413 cities would cost US\$ 3,717,000.

A summary of all costs involved in this estimation is presented in Table 4. All figures in the budget calculations are in US dollars.

Table 4: Summary of costs over four years.

Description	Cost – US\$	
Radosys Unit		23,736
Detectors acquisition (434,087 units)		2,094,037
Transport of detectors		46,540
Detectors distribution logistic		3,717,000
Sub-total (all above): Direct cost	5,881,313	
Travel for team members & meetings		120,000
Training (postdocs)		160,000
Grand Total		6,161,313

5. Conclusion

The results of the survey will allow to conduct exposure assessment for the population in South America and to assess the health impact of this exposure. The results of the project will help national health authorities in developing national residential radon action levels and regulations, as well as provide public health guidance for awareness-raising and mitigation.

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Remmeters Calibration at Neutron Calibration Center of Nuclear Regulatory Authority

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Remmeters calibration at Neutron Calibration Center of Nuclear Regulatory Authority

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Abstract. This paper presents the calibration of remmeters with two different methods following the ISO8529 requirements. The aim is to obtain the dose equivalent response defined as the relationship between the measurement of a rate and the Ambiental Dose Equivalent, $H^*(10)$.

The first method applied is the well-known Generalized –fit method (GFM) of ISO 8529:2000 (E).

The second one is based on the knowledge of the value of $H^*(10)$ rate in several points of the calibration bank. The value of this quantity was acquired in two different ways: a) measuring with the Rospec II transfer detector calibrated by PTB (Germany) in the thermal region and by the NIST (USA) up to 20 MeV; and b) calculating with the MCNPX code, considering the actual facility (dimension, walls and bank materials, sources).

The calibration of neutron detectors was carried out in the ARN's Neutron Calibration Center, with ^{252}Cf bare and ^{252}Cf moderated with D_2O sources. The detectors selected were LB6411 (Berthold) and NM2B (Thermo Scientific). As the 6 m long calibration bank, two scales were tested in order to check the linearity.

Several points stood out from these calibrations:

a) As the neutron scattering effects of the NCC are not very important (negligible, about 3% at 70 cm from the source) the generated fit function suggested by ISO is over designed for the corrections needed in this facility.

b) The acquisition times must be defined for each scale and it would be interesting to include them in the calibration report.

c) The GFM parameters of the detectors do not depend on the central - detector geometry.

The group of parameters presented in this work could be considered as a guide by other laboratories to adjust their algorithms even though the parameters depend on the detector, the sources and the facility itself.

1. Introduction

To calibrate is to find the relationship between the reading of an instrument and the conventionally true value of the quantity to be measured [1]. In the case of neutron detectors calibration it is necessary to consider not only the neutron coming from the source but also the neutrons scattering by the neighboring area, walls, ceilings, calibration bank, air, detector itself and so on. The fluence as well the spectral fluence of the neutron scattering field depends on the detector position in relation to the source. A typical behavior is the increase of the proportion of thermal neutrons with respect to the total neutrons with the distance to the source. In case of our laboratory the proportion is 1:4:10:17 for ^{252}Cf and 1:2:3:5 for ^{252}Cf moderated at 70 cm: 150 cm: 300 cm: 500 cm respectively to the source [2].

The active detectors based on ^3He or F_3B and even passive detectors as the TL crystals have a natural tendency to follow the cross section of interaction, which is grown-up in the thermal region.

The detectors have different responses at different positions along the calibration track motivated not only by the expected variation in the dose rate but also in the variation in the quality of radiation that is produced. In order to solve this, ISO 8529 recommends measuring once certain number of device readings at different centers-source to center-detector (c-c) distances and fitting them into a curve. Applying the ISO-Generalized Fit Method, the parameters obtained can correct the deviation of the actual device readings from the simple inverse-square law (followed by an ideal experimental arrangement). This set of parameters allows calibrating a device measuring only once c-c distance. Other c-c distances could be measured in order to test the detector performance at different dose rates.

The second method applied is based on the conventionally true value of the Ambiental dose equivalent rate [3] at several c-c distance of the calibration bank. The value of this quantity was obtained in two different ways: a) measuring with the transfer detector, Rospec II¹ [4], and b) calculating with

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MCNPX [5] code, considering the actual facility (dimension, walls and bank materials, sources and so on). The value assigned to each point was the average of MCNPX+ROSPEC.

2. The Irradiation Facility

Calibrations were made in the Neutron Calibration Center of the Nuclear Regulatory Authority of Argentina (ARN in Spanish). The laboratory consists of an elevated track system made of aluminum for positioning detectors above the irradiator for calibration and two neutron fields, ^{252}Cf bare and moderated with D_2O . The irradiator permits the user to perform irradiations automatically from a remote location in the control room.

2.1 Neutron source characteristics

The neutron source is ^{252}Cf housed in a type 100NS capsule, manufactured by Frontier Technology. The mass of ^{252}Cf reported by the manufacturer is stated as 121.4 μgr with a reference date of May 10, 2005. The conversion factor used was $2.34 \cdot 10^{12}$ n/seg-gr [6]. The source was measured with and without a 30-cm D_2O moderator sphere covered by 1mm of cadmium. The anisotropy factor for ^{252}Cf nude was 1.19 and 1.0 for $^{252}\text{Cf}(\text{D}_2\text{O}, \text{Cd})$ [2].

3. Methodology

Calibration of the two neutron detectors applying two different methodologies was carried out. The proposed methodology was ISO-8529 and the Los Alamos National Laboratory (LANL) reference value.

The neutron detectors were:

Berthold LB6411+LB1230, a spherical device with a detection limit of 30 nSv/h. [7].

Thermo Scientific NM2B, a cylindrical device with a detection limit of 1 $\mu\text{Sv/h}$.

3.1 ISO 8529-3 methodology

The method proposed by the ISO 8529 was applied to each remmeters for both sources. The device readings were taken at different positions along the 6m-long calibration bank. In each position, the device was read five times in order to calculate the average and its dispersion. The measured quantity, the Ambient Dose Equivalent Rate, $H^*(10)$ was plotted as a function of the c-c distance to analyze the data goodness. Then, the product between the square c-c distance and $H^*(10)$ is plotted as a function of the c-c distance in order to obtain the first “guess” for the parameters.

These parameter values (order zero parameters) were included in the ISO-algorithm (**Generalized Fit Method (GFM)**) and each term behavior was analyzed as a function of c-c distance. The procedure applied to obtain the set of parameters was the minimization of G,

$$G = \frac{1}{N} \sum [H^*(10)_{\text{measured}} - H^*(10)_{\text{calculated}}]^2$$

where N is the number of measurements performed

The background measurements were negligible.

Later, the goodness of the set of parameter was checked again, by plotting the residuals of the fitting as well as the quotient as a function of c-c distance.

The ISO formalism was applied to each set remmeter - source.

a1) LB6411 with ^{252}Cf bare,

$$F_1(d) = 1 + a_4 * (1 + a_5 * (d/12.5 - 1))^{-2}$$

$$c(d) = K * (F_1(d) * \exp(-d * 1055 \cdot 10^{-7}) / d^2 + A/d + S)$$

a2) LB6411 with ^{252}Cf (D_2O , Cd),

$$F_1(d) = 1 + a_4 * (1 + a_5 * (d/12.5 - 2.2))^{-2}$$

$$c(d) = K * (F_1(d) * \exp(-d * 2964 * 10^{-7}) / d^2 + A/d + S)$$

b1) NM2B with ^{252}Cf bare,

$$F_1(d) = 1 + a_4 * (1 + a_5 * (d/10.75 - 1))^{-2}$$

$$c(d) = K * (F_1(d) * \exp(-d * 1055 * 10^{-7}) / d^2 + A/d + S)$$

b2) NM2B with ^{252}Cf (D_2O , Cd),

$$F_1(d) = 1 + a_4 * (1 + a_5 * (d/10.75 - 2.506))^{-2}$$

$$c(d) = K * (F_1(d) * \exp(-d * 2964 * 10^{-7}) / d^2 + A/d + S)$$

Table 1 shows the set of parameters obtained for LB6411 and NM2B detectors in the neutron fields (^{252}Cf and ^{252}Cf (D_2O , Cd)) of the NCC facility.

Table 1. Parameters obtained applying ISO methodology for LB6411 and NM2B in the NCC neutron fields

Parameter	LB6411		NM2B	
	^{252}Cf	^{252}Cf (D_2O , Cd)	^{252}Cf	^{252}Cf (D_2O , Cd)
A [cm^{-1}]	4.290E-04	2.620E-03	7.481E-04	5.395E-04
S [cm^{-2}]	7.561E-07	-2.400E-06	7.746E-08	1.755E-06
a ₄	-2.233	0.02002	4.679E-08	-5.896E-03
a ₅	1.790	-1.233	0.015	1.146
K [$\text{mSv cm}^2/\text{h}$]	26269	6024	21374	6781
χ^2 [mSv/h]	1.7E-03	4.3E-05	1.2E-03	8.5E-05

For the calibration purpose, the most important parameter is K, because it represents the neutron angular source strength without any backscatter contribution, so the response is obtained from K, dividing it by the neutron angular source strength.

Other laboratories could consider the auxiliary parameters presented as a guide to adjust algorithms even though the parameters depend on the detector, the sources and the facility itself. This sort of detector response is a free field response, that is to say, the response is independent of the laboratory. Moreover, this is consistent with the ISO philosophy: “the response or calibration factor of a device is a unique property of the type of device, and may depend on dose-equivalent rate, the neutron source spectrum or the angle of incidence of the neutrons, but should not be a function of the characteristics of the calibration facility or the experimental techniques used” [1].

3.2 Applying the reference value

As the laboratory was characterized by LANL with a ROSPEC II transfer detector as well as the MCNPX code [2], the Ambiental Dose Equivalent rate values reported at different positions of the calibration bank were considered as reference. In this case, the response was defined as,

$$R = \frac{\text{Instrument reading}}{\text{reference value}} = \frac{H * (10)_{\text{measured}}}{H * (10)_{\text{reference}}}$$

4. Calibration

The responses of neutron devices NM2B and LB6411 are presented in Table 2 and Table 3 respectively, according to the two methods previously explained.

Table 2. Response factors of the NM2B detector as a function of c-c distance.

Source	Distance	Response (LANL reference value)	Response ISO 8529-3
^{252}Cf bare	70 cm	1.09	1.05
	300 cm	1.07	
^{252}Cf (D_2O , Cd)	70 cm	1.25	1.65
	300 cm	1.22	

Table 3. Response factors of the LB6411 detector as a function of c-c distance.

Source	Distance	Response (LANL reference value)	Response ISO 8529-3
^{252}Cf bare	70 cm	1.05	1.04
	300 cm	1.02	
^{252}Cf (D_2O , Cd)	70 cm	1.02	1.16
	300 cm	1.02	

5. Discussion

As the neutron scattering effects of the NCC are negligible (less than 3% at 70 cm from the source), the generated fit function (GFF) suggested by GFM of ISO is over designed for the corrections needed in this facility.

The calibration process measures the c-c distance (d_{measured}) and the dosimetry quantity at d_{measured} , $H^*(10)$. Therefore, it is necessary to control that not systematic errors occur in the positioning.

The procedure carried out to test the occurrence of these systematic errors is:

- 1) Calculate the distance, ($d_{\text{calculated}}$) so that $\text{GFF}(d_{\text{calculated}})$ is equal to $H^*(10)_{\text{measured}}$
- 2) $d_{\text{calculated}} - d_{\text{measured}}$ is calculated
- 3) $d_{\text{calculated}} - d_{\text{measured}}$ as a function of d_{measured} are plotted.

In all of our studied cases, the resulted plotting shows a random distribution centered on zero.

The acquisition times of each detector must be defined for each scale and it would be interesting to include this value in the calibration report.

GFM parameters of the device do not depend on the central - detector geometry.

The NM2B presented a special behavior close to the rate dose detection limit and it is necessary to spend longer (15 min) to achieve readings stabilization.

9. Conclusions

The calibration of two different neutron devices was carried out in the NCC applying the ISO 8529 procedure and LANL reference value. The work done is the starting point to establish the standard operating procedure of the ARN laboratory.

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Bonner Sphere Spectrometer: a CONRAD Project Intercomparison

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Bonner Sphere Spectrometer: A CONRAD Project Intercomparison

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Abstract. The most widely used system in neutrons measurements for radiological protection is the Bonner Sphere Spectrometer (BSS). The BSS is applied to characterise neutron fields from thermal to hundreds of MeVs. The Nuclear Regulatory Authority of Argentina has developed and calibrated its own BSS system, which has been used in many Argentine facilities during the last eleven years when the regulatory activities have been carried out. Following this line of work, the present development has been done in the framework of the International Intercomparison "Uncertainty Assessment in Computational Dosimetry: A Comparison of Approaches", organised by the CONRAD project (Coordinated Network for Radiation Dosimetry). The aim of intercomparison was to study the response of a proposed widespread neutron spectrometer exposed to arbitrary neutron sources. With this goal in mind, the experimental system has been modelled in detail according to the provided layout. The modelled neutron spectrometer consists of 8 Bonner spheres made of high-density polyethylene ($\delta=0.95\text{gc/m}^3$). The spheres diameter range between 2" and 12" in addition to a 12" diameter lead-loaded sphere. The defined active thermal neutron detector, a ${}^6\text{LiI(Eu)}$ scintillation crystal, was according to provided dimensions (4 mm (diameter) by 4 mm (height)), and located at each sphere centre. Irradiation geometry has been according to measurements carried out during the experimental part of the intercomparison. The theoretical neutron response has been calculated applying the well-known MCNPX code. The complete response matrix of the system has been obtained in the energy range between thermal neutron and 17.77 MeV. The obtained system theoretical response to ISO standard ${}^{241}\text{Am-Be}$ and ${}^{252}\text{Cf}$ sources shows an excellent agreement with experimental results provided by EURADOS. This response can be used to calibrate the system. The obtained matrix response can be coupled to any unfolding code to complete the BSS system used in the intercomparison or similar.

KEYWORDS: *Bonner Spheres, Neutron spectrometry, Scintillator detectors, Monte Carlo simulation*

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1. Introduction

The most widely used system in neutrons measurements related to radiological protection is the Bonner sphere spectrometer (BSS)[1-2]. It is based on a thermal detector, a set of polyethylene spheres, and the associated electronics in the case of an active detector like $^{10}\text{BF}_3$ or ^3He , or scintillators like $^6\text{Li}(\text{Eu})$. The BSS is applied to characterise the neutron field from thermal to hundreds of MeVs. One of the main advantages of the BSS system is its isotropic response to neutrons resulting from its spherical symmetric shape.

The BSS delivers a good performance for radiation protection purposes, in spite of its low resolution, but a net limitation appears in the case of active detectors in either pulsed or high intensity environmental radiation fields because of pile up effects or undesirable dead times. The best solution for this problem is to use passive detectors (activation, track and thermoluminescent), which have been applied in aircrew dosimetric determinations, and pulsed fields from medical and research accelerators.

Once experimental responses are known, an unfolding code can be used to get the neutron spectrum incident on the detector. In this way, the system is used to evaluate any arbitrary incident neutron spectrum from environment, or generated by a given radiation facility. Once the neutron spectrum is obtained, using recommended conversion coefficients, integration may give the desired radiological quantity [3].

Every unfolding code needs basically two user inputs: an initial approximate neutron spectrum, and the detector response as a function of incident neutron energy (for the full set of spheres). This complex function is called response matrix. As various investigations have shown [4], a good knowledge of the response matrix of the BSS is crucial to obtain reliable spectrometric results. There are two approaches to determine the response matrix: experimental calibration in monoenergetic neutron beams and neutron transport calculations. Experimental determination is limited by a series of factors: the small number of available monoenergetic neutron beams under calibrated field conditions, the limited energy range where neutron fields may be generated, the room scattering of neutrons which complicate the measurement of direct neutron component and, sometimes, the poor counting statistics. Because of these problems, usually a theoretical determination is used, applying a Monte Carlo simulation.

The Nuclear Regulatory Authority of Argentina has developed and calibrated its own BSS system, which has been used and improved by the experience gained in many Argentine facilities during the last eleven years while carrying out regulatory activities. Following this line of work, the study of the neutron response of a proposed BSS equipped with a $^6\text{Li}(\text{Eu})$ was undertaken. This study was done in the framework of the International Intercomparison "Uncertainty Assessment in Computational Dosimetry: A Comparison of Approaches" organised by the CONRAD project (Coordinated Network for Radiation Dosimetry) which is coordinated by the European Radiation Dosimetry Group (EURADOS). The aim of the intercomparison was to study the theoretical and experimental response of a proposed widespread neutron spectrometer exposed to arbitrary neutron sources. This work comprises the theoretical study of responses and subsequent comparison with experimental data provided by EURADOS.

2. An important precedent: The Nuclear Regulatory Authority (ARN in Spanish) BSS system

The ARN's experimental development began in 1995. Later, that work was accompanied by an intensive theoretical task to model the system response. That joint effort gave rise to a complete BSS system (with its own response matrix), which is continuously being updated. This BSS was used to get the neutron spectra of a variety of Argentine facilities: experimental reactors, linear accelerators, cyclotrons, and medical accelerators [5-6-7].

The system consists mainly of a set of high-density (about 0.95 g/cm^3) polyethylene spheres with the following diameters: 0"(bare detector), 3", 4", 5", 6", 7", 8", 10", and 12", and a spherical (32 mm

diameter) gaseous ^3He detector (SP9 from Centronic), which is used as central active detector. Additionally, a cadmium cover is used for 0", 3", 4" and 5" spheres. Pulse acquisition is carried out by an EG Ortec Nomad Plus multichannel equipment, coupled to an EG Ortec 142PC preamplifier. The whole system is connected to a portable PC with the adequate multichannel analysing software. Neutron spectra are obtained deconvoluting experimental responses by means of UMG33 unfolding software [8], which is based on MAXED code.

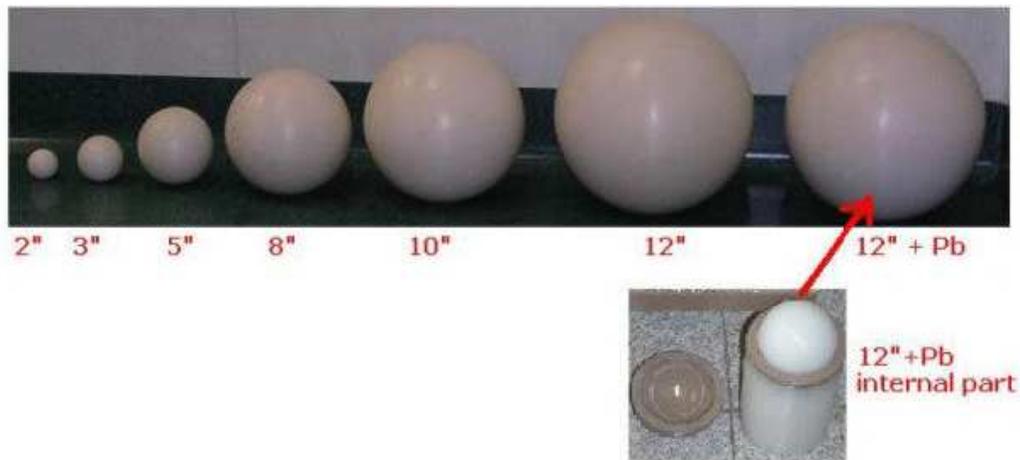
The response matrix (used by the software) was obtained using the MCNPX Monte Carlo code [9]. This code can transport neutrons (and other particles) up to hundreds of MeVs, but in this case it was used in the range from thermal to 100 MeV. The full capability of the code might be used in case of detectors response to cosmic radiation, which is of interest in radiation protection applied to commercial aviation. The set of spheres was calibrated (both diameter and density) at the Industrial Technology National Institute of Argentina (INTI in Spanish). On the other hand, the whole system was calibrated by L'Institut de Radioprotection et de Sûreté Nucléaire (IRSN) in the Cadarache Centre, France.

As mentioned above, active systems have some limitations. In order to overcome these difficulties, ARN began to develop a BSS system with passive detectors, specifically with TLD600 thermoluminescent dosimeters (i.e. $^6\text{LiF:Mg,Ti}$) [10-11]. This BSS system is currently under development.

3. Intercomparison 's experimental system description

This BSS consists of a set of 0.95 g/cm^3 density polyethylene spheres. The spheres diameters are: 0" (bare detector), 2", 3", 5", 8", 10" and 12". Such configuration allows to get spectral information from thermal to 20 MeV neutrons. In order to extend the energy range up to hundreds of MeVs, to a new 12" diameter sphere a 1-cm lead layer is added at 8 cm from the centre. This new configuration profits from the production of (n,xn) reactions induced by high energy neutrons in the lead shell. Fig 1 shows the complete set, with details of the lead-loaded sphere.

Figure 1: The set of spheres: 2", 3", 5", 8", 10", 12" and 12"+Pb with details of internal configuration

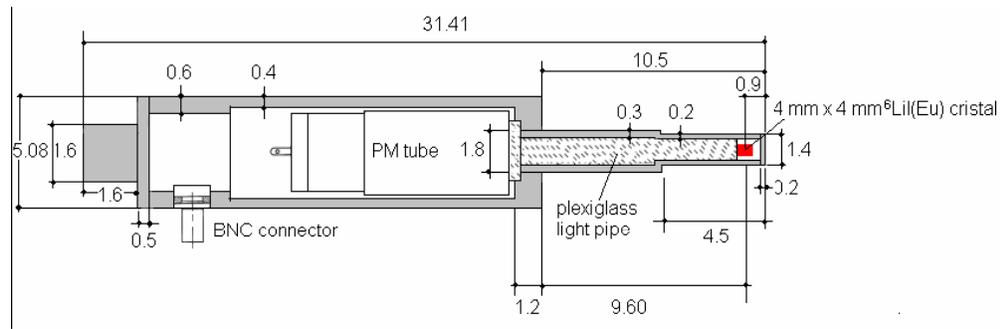


A cylindrical 4 mm (diameter) by 4 mm (height) $^6\text{LiI(Eu)}$ (96% of ^6Li) active scintillator is used as central detector. Thermal neutrons are detected by the $^6\text{Li}(n,\alpha)^3\text{H}$ reaction (Q value = 4.78 MeV). A scaler counts the electronic pulses from the photomultiplier. The number of $^6\text{Li}(n,\alpha)^3\text{H}$ reactions in the scintillator is considered to be proportional to the number of pulses registered by the scaler; the photon sensitivity of the detector is supposed to be negligible.

A cross sectional view of the central detector is shown in Fig 2. The whole assembly has a cylindrical geometry. The active part is labelled as "4x4 mm $^6\text{Li(Eu)}$ crystal", and its centre is located at 0.9 cm

from the right external surface of the Aluminium assembly. Grey represents aluminium, white is vacuum and the light pipe connecting the scintillator to the photomultiplier is Plexiglas.

Figure 2: Layout of the central detector of the BSS. All dimensions are in cm.



4. Calculation procedure

The programme used in the calculations is the MCNPX Monte Carlo code, version 2.6d, developed at Los Alamos National Laboratory [12].

As MCNPX works with three dimensional geometry configurations, the experimental system was modelled in detail according to the provided detector layout. The only variance reduction technique used to minimise stochastic uncertainties, was geometry splitting/Russian roulette (in the bigger spheres). All the spheres were considered to be exposed to a parallel neutron beam of the same cross sectional area of each sphere. The irradiation geometry was such that the detector axis was parallel to the radiation field with the light pipe opposing the field direction. Furthermore, all irradiations were considered in vacuo.

5. Cross section libraries

The programme cross section treatment is continuous, with linear interpolation between energy points. In this work, energy varied between thermal and 17.776 MeV, as required for the intercomparison. The continuous energy neutron cross sections, available in the standard MCNPX package, were from Evaluated Nuclear Data File (ENDF) [13] and from Lawrence Livermore National Laboratory (LLNL) [14] libraries, according to the following detail:

- ${}^6\text{Li}$ from ENDF/B-VI release 1.
- ${}^7\text{Li}$ from ENDF/B-VI release 0.
- ${}^1\text{H}$ and C (natural) from ENDF/B-VI release 6.
- ${}^{27}\text{Al}$ from ENDF/B-VI release 8.
- ${}^{153}\text{I}$ and Pb (natural) from LLNL.

For neutron transport the free gas model down to 4 eV is used and for lower energies the $S(\alpha,\beta)$ treatment takes into account the crystalline structure of materials for thermal neutrons interactions. In this particular case the following library was used:

- poly.60t (hydrogen in polyethylene at 294 degrees Kelvin) from ENDF/B-VI release 3, for the case of polyethylene and PMMA (polymethyl methacrylate, for modelling Plexiglas).

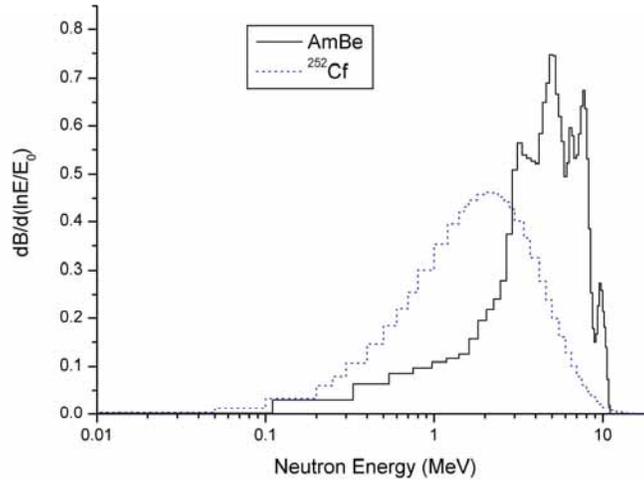
6. Computed quantities

The basic result (or “tally”) obtained by MCNPX is the neutron fluence energy distribution within the scintillator detector. Using adequate cross sections, the number of ${}^6\text{Li}(n,\alpha){}^3\text{H}$ reactions per incident neutron can be obtained. In this exercise, the response was mainly calculated in terms of the number of ${}^6\text{Li}(n,\alpha){}^3\text{H}$ reactions per unit incident fluence (i.e. cm^2 as unit) expressed as a function of neutron energy.

7. Source Energy Distributions

The neutron sources studied were monoenergetic neutrons and neutron reference spectra. The neutron reference spectra investigated were bare $^{241}\text{Am-Be}$ and ^{252}Cf . The spectra for these sources were taken from the standard ISO 8529-1[15], and they are shown in Fig 3, where B (expressed in lethargic form) is the source strength of neutrons with energies between E and E+dE.

Figure 3: Lethargic representation of the neutron spectra of ^{252}Cf and Am-Be sources according to ISO 8529-1

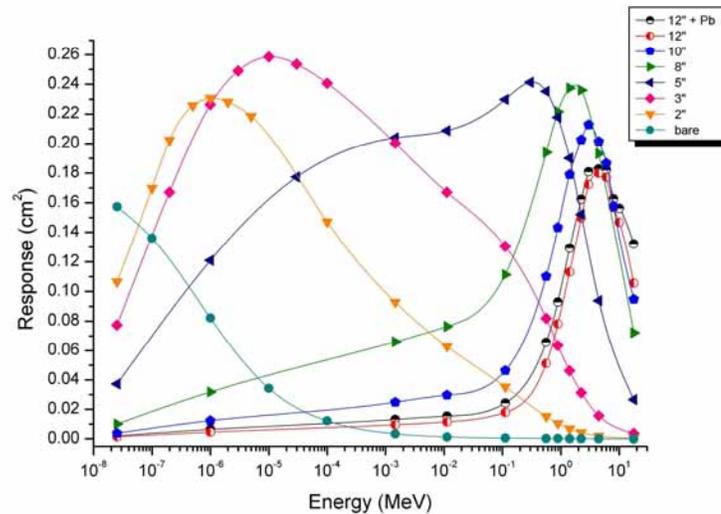


8. Theoretical results

8.1 Response matrix of the BSS system

The response of the BSS proposed was studied for monoenergetic neutron beams. Fig 4 shows the obtained response matrix of the BSS system expressed as the number of $^6\text{Li}(n,\alpha)^3\text{H}$ reactions normalised to unit incident fluence (cm^2).

Figure 4: Response matrix of the BSS system in the range from thermal to 17.776 MeV



8.2 Eight-inch sphere energy response

The particular case of the 8" sphere response was studied. Results are shown in Table 1. The detector response is expressed in two forms: the number of ${}^6\text{Li}(n,\alpha){}^3\text{H}$ reactions normalised to unit incident fluence (response in cm^2), and the sphere response normalised to the maximum value (relative response).

Table 1: Energy response of the 8" sphere

Energy (MeV)	Relative response to maximum	Standard deviation	Response (cm^2)	Standard deviation (cm^2)
$2.53 \cdot 10^{-8}$	0.042	0.001	$1.00421 \cdot 10^{-2}$	$1.46614 \cdot 10^{-4}$
$1.00 \cdot 10^{-6}$	0.134	0.003	$3.17846 \cdot 10^{-2}$	$4.32270 \cdot 10^{-4}$
0.00147	0.278	0.005	$6.59352 \cdot 10^{-2}$	$7.38474 \cdot 10^{-4}$
0.01123	0.321	0.006	$7.63385 \cdot 10^{-2}$	$8.47358 \cdot 10^{-4}$
0.11225	0.469	0.009	$1.11388 \cdot 10^{-1}$	$1.11388 \cdot 10^{-3}$
0.56226	0.817	0.014	$1.93998 \cdot 10^{-1}$	$1.76539 \cdot 10^{-3}$
0.89107	0.933	0.016	$2.21580 \cdot 10^{-1}$	$1.83911 \cdot 10^{-3}$
1.41100	1.000	0.017	$2.37486 \cdot 10^{-1}$	$2.01863 \cdot 10^{-3}$
2.24000	0.994	0.014	$2.36118 \cdot 10^{-1}$	$1.36948 \cdot 10^{-3}$
4.46540	0.813	0.012	$1.93146 \cdot 10^{-1}$	$1.15887 \cdot 10^{-3}$
17.7760	0.304	0.004	$7.22038 \cdot 10^{-2}$	$3.39358 \cdot 10^{-4}$

8.3 BSS response to ${}^{241}\text{Am}$ -Be source

The response of the BSS exposed to a bare ${}^{241}\text{Am}$ -Be source was studied. Results are shown in Table 2 and Fig 5. The detector response is expressed in the same form as in the previous case, but in this case relative response means that response is normalised to 8" sphere response.

Table 2: Response of the BSS system to ${}^{241}\text{Am}$ -Be source

Sphere diameter (inches)	Relative response to 8"	Standard deviation	Response (cm^2)	Standard deviation (cm^2)
2	0.027	0.0003	$5.05127 \cdot 10^{-3}$	$3.28332 \cdot 10^{-5}$
3	0.159	0.002	$2.94312 \cdot 10^{-2}$	$1.47156 \cdot 10^{-4}$
5	0.646	0.005	$1.19383 \cdot 10^{-1}$	$2.62642 \cdot 10^{-4}$
8	1.000	0.010	$1.84691 \cdot 10^{-1}$	$9.41925 \cdot 10^{-4}$
10	0.941	0.012	$1.73831 \cdot 10^{-1}$	$1.30373 \cdot 10^{-3}$
12	0.790	0.009	$1.45903 \cdot 10^{-1}$	$8.60825 \cdot 10^{-4}$

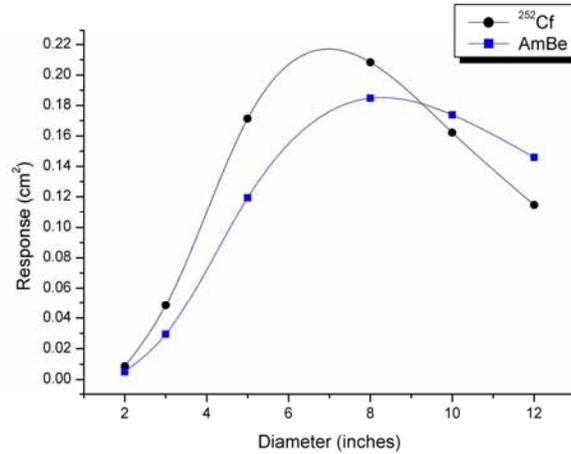
8.4 BSS response to ${}^{252}\text{Cf}$ source

The response of the BSS exposed to a bare ${}^{252}\text{Cf}$ source was studied. Results are shown in Table 3 and Fig 5. The detector response is expressed in the same form as in the previous case.

Table 3: Response of the BSS system to ${}^{252}\text{Cf}$ source

Sphere diameter (inches)	Relative response to 8"	Standard deviation	Response (cm^2)	Standard deviation (cm^2)
2	0.041	0.0004	$8.55018 \cdot 10^{-3}$	$4.61710 \cdot 10^{-5}$
3	0.232	0.002	$4.83536 \cdot 10^{-2}$	$2.22426 \cdot 10^{-4}$
5	0.822	0.010	$1.71267 \cdot 10^{-1}$	$1.16462 \cdot 10^{-3}$
8	1.000	0.011	$2.08345 \cdot 10^{-1}$	$1.14590 \cdot 10^{-3}$
10	0.779	0.011	$1.62227 \cdot 10^{-1}$	$1.39516 \cdot 10^{-3}$
12	0.551	0.007	$1.14695 \cdot 10^{-1}$	$8.48743 \cdot 10^{-4}$

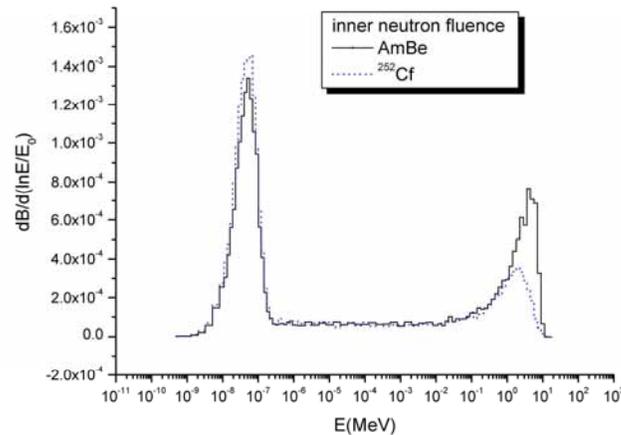
Figure 5: Response of the BSS system to ^{252}Cf and Am-Be sources, as a function of sphere diameter.



8.5 Study of neutron spectra at the entrance of the $^6\text{Li}(\text{Eu})$ detector

Although the knowledge of the neutron spectrum at the centre of the sphere is not mandatory for a good management of the Bonner Spheres technique, it helps to understand the “spectrum shifter” action of the polyethylene moderator. The study of both $^{241}\text{Am-Be}$ and ^{252}Cf sources indicate why the response of a given sphere changes as the incident neutron spectrum changes. The neutron spectrum was determined within an inner thin lenticular layer (0.1 mm thick) inside the 8” sphere exposed to $^{241}\text{Am-Be}$ and ^{252}Cf sources, whose normal surface is parallel and opposite to the beam source. The results are shown in Fig 6.

Figure 6: Neutron fluence incident in the ^6Li detector, for ^{252}Cf and Am-Be sources



9. Experimental results

Experimental results provided by EURADOS [16] are presented and compared with the theoretical ones of this work. The experimental response of the BSS exposed to a bare $^{241}\text{Am-Be}$ source is shown in Table 4 and Fig 7. The detector response is expressed normalised to 8” sphere response. In the same way, Table 5 and fig 8 show results for ^{252}Cf source.

Table 4: Experimental response of the BSS system to $^{241}\text{Am-Be}$ source, relative to 8" sphere

Sphere Diameter (inches)	Relative Response	Standard Deviation
2	0.032	0.0006
3	0.176	0.002
5	0.659	0.007
8	1.000	0.020
10	0.947	0.019
12	0.793	0.016

Table 5: Experimental response of the BSS system to ^{252}Cf source, relative to 8" sphere

Sphere Diameter (inches)	Relative Response	Standard Deviation
2	0.046	0.0009
3	0.238	0.002
5	0.874	0.017
8	1.000	0.020
10	0.811	0.016
12	0.557	0.011

Figure 7: Comparison between theoretical and experimental responses of the BSS system to Am-Be source, as a function of sphere diameter, relative to 8" sphere

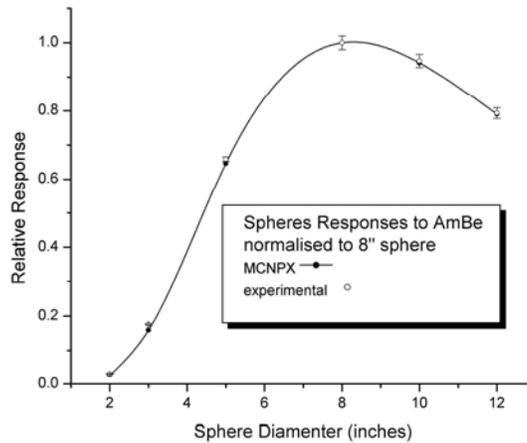
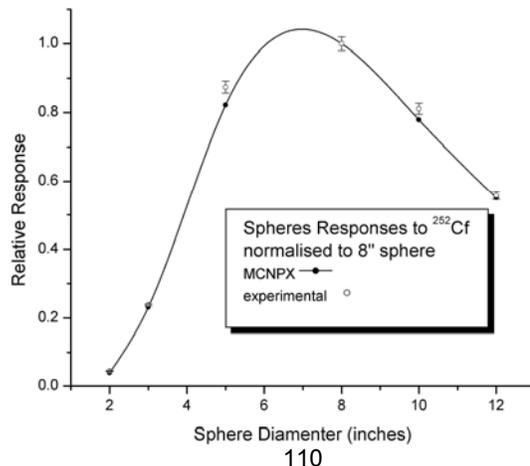


Figure 8: Comparison between theoretical and experimental responses of the BSS system to ^{252}Cf source, as a function of sphere diameter, relative to 8" sphere



10. Conclusions

The neutron response of the proposed BSS was calculated both for monoenergetic wide energy range beams and for standard ISO Am-Be and ^{252}Cf sources.

Although the intercomparison programme required only the 8" sphere neutron response to monoenergetic beams from thermal to 17.77 MeV, the complete BSS matrix response (in that energy range) was calculated and presented in this work. This goal was achieved because of the importance of response matrix in BSS applications. By using the BSS neutron response matrix and the neutron spectrum of any arbitrary source, it is possible to calculate, by integration, the response of each sphere in order to calibrate the system. The obtained matrix response can be coupled to any unfolding code to complete the BSS system corresponding to the $^6\text{LiI}(\text{Eu})$ detector used in the intercomparison.

Theoretical results show a very good agreement with experimental measurements provided by EURADOS (Fig 7 and 8). This result is a direct experimental validation of the computational model used at this work. This outcome is of considerable importance because the response of the system to standard $^{241}\text{Am-Be}$ and ^{252}Cf sources may be used to calibrate the system.

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Bonner Sphere Spectrometry: Main Achievements from the “CONRAD Computational Dosimetry Study”

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Bonner Sphere Spectrometry: main achievements from the “CONRAD Computational Dosimetry Study”

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Abstract. In the framework of CONRAD-Work Package 4 on Computational Dosimetry a series of problems was proposed concerning the usage of computer codes as supporting tools for irradiation facilities calibration procedures or detector characterization in the field of radiation protection. In this paper the results submitted by the various laboratories for the Bonner Sphere Spectrometer response characterization problem (one of the proposed studies) are summarized. The aim of this problem is to study the response of a widespread neutron spectrometer exposed to the ISO standard neutron sources ²⁴¹Am-Be and ²⁵²Cf.

The Bonner Sphere Spectrometer (BSS) is a widely used system to evaluate the neutron spectrum through a series of different diameter moderating spheres (in this case polyethylene) including active or passive detectors. An accurate knowledge of the energy dependence of the response of each sphere is necessary to reconstruct the spectrum from the measurement (counts or activities) through an “unfolding” procedure.

In the proposed comparison study, a typical BSS is proposed. The sphere diameters are: 0” (bare detector), 2”, 3”, 5”, 8”, 10” and 12”, such configuration allows getting spectral information from thermal to 20 MeV neutrons. The BSS were exposed to calibrated ISO sources at INFN and ENEA laboratories. The employed detector was a LiI scintillator.

Participants to the intercomparison were asked to determine the response of a minimal set of BS exposed to a parallel neutron beam for ²⁵²Cf and ²⁴¹Am-Be sources; simulate the irradiation of the 8” sphere to a parallel neutron beam for a set of given neutron energies and determine the neutron spectrum within the inner thin lenticular layer (0.1 mm thick) inside the Bonner sphere towards the source and perpendicular to the beam axis. The last point was important to check the correct modelling of the neutron slowing down within the spheres before getting to the scintillator, as it helps to clearly understand the “spectrum shifter” action of the polyethylene moderator. Fourteen solutions were submitted and discussed during the CONRAD Workshop “Uncertainty Assessment in Computational Dosimetry: A comparison of approaches”, held in Bologna (Italy) from October 8th to 10th 2007.

In general the majority of the solutions were fitting quite satisfactorily with the authors’ reference solutions. Some outliers were nevertheless detected. The main causes of the deviations could be ascribed to an inappropriate sampling of the source both in space and energy, to the erroneous treatment of hydrogen and carbon elastic scattering cross-sections as “free gas” instead of taking into account the binding of the nuclei in the polyethylene molecule, and some mistakes in the interpretation of the units in which to express the absolute response function

KEYWORDS Monte Carlo, Bonner Spheres, Neutron Spectrometry

1. Introduction

In the framework of the EU Coordinated Action CONRAD (Coordinated Network for Radiation Dosimetry) within Working Package 4 (WP4), the aim of the proposed problem dealing with “Bonner Sphere Spectrometry” was to assess the ability of the various Monte Carlo users to correctly model the sphere system response and compare the calculated values with experimental results obtained in well defined neutron field conditions, i. e. the irradiation with ^{252}Cf and $^{241}\text{Am-Be}$ ISO reference sources in a characterized neutron irradiation room. The study was subdivided in three main steps:

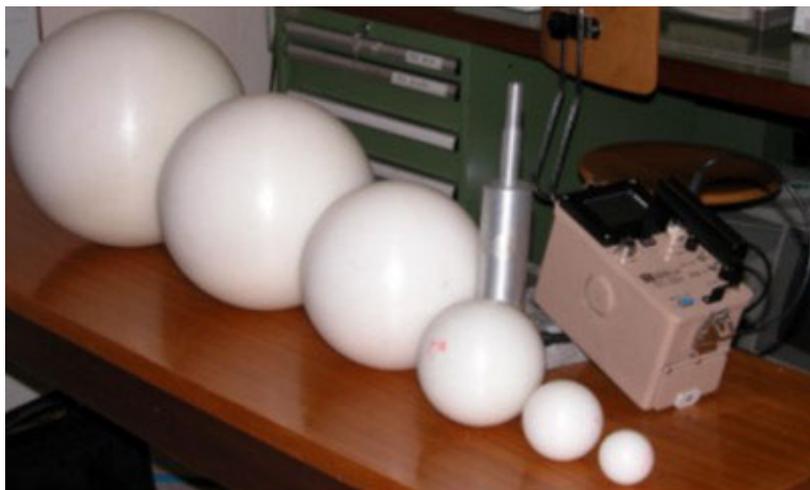
- The study of the energy response function for the 8” sphere to be compared with literature results (“step zero” to check the consistency of the procedure).
- The response calculation for the full set of spheres of different diameters irradiated with ^{252}Cf and $^{241}\text{Am-Be}$ sources and the comparison with measured data
- The evaluation of the incoming neutron spectra (^{252}Cf and $^{241}\text{Am-Be}$ sources) at the entrance of the neutron detector inside the measurement cavity of the Bonner sphere. The scope of this study was to state the correct way to model the neutron slowing down and thermalization process within the moderating material, which strongly influences the detector response.

2. Materials and methods

2.1 Moderation and detection: slowing down in infinite, homogeneous, non multiplying media

The instrument is the multi-sphere or Bonner Sphere Spectrometer (BSS), firstly introduced by Bramblett et al [1] in the 60s. The spectrometer is constituted by a set of polyethylene spheres, usually in the order of 6 to 10, with known diameter and density, with an active or passive thermal neutron sensor located at their centre (in Figure 1 the BSS studied in this work is presented). Since each sphere preferably moderates the neutrons in a given energy range, the set of the measurements obtained from the different spheres will provide information on the energy distribution of the incident neutrons. Provided an accurate knowledge of the energy dependence of the response of each sphere, $R_i(E)$ (where “i” indicates the sphere and E the neutron energy), an “unfolding” computer code is used to estimate the neutron spectrum. This is usually done by changing iteratively an initial “guess” spectrum, till the correspondent calculated Bonner sphere responses are in agreement with the measurements.

Figure 1. The set of spheres: 2”, 3”, 5”, 8”, 10” and 12”. The detector and the related data logger are also included.

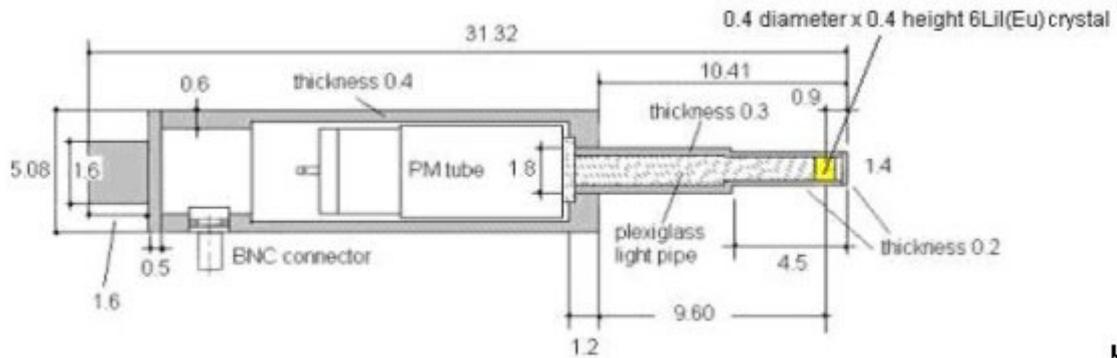


2.2 Description of the instrumentation and data used for the simulation

A cylindrical 4 mm (diameter) x 4 mm (height) $^6\text{Li(Eu)}$ (96% of ^6Li) active scintillator is used as central detector. Thermal neutrons are detected through the $^6\text{Li}(n,\alpha)^3\text{H}$ reaction (Q value = 4.78 MeV).

The electronic pulses from the photomultiplier, after a pulse-height analysis, are counted by a scaler. The photon sensitivity of the detector is assumed to be negligible. A cross sectional view of the central detector is shown in Figure 2. The whole assembly has a cylindrical geometry. The active part is labelled as “4x4 mm ${}^6\text{Li}(\text{Eu})$ crystal”, and its centre is located at 0.9 cm from the right external surface of the Aluminium assembly (in figure all parts in grey are in aluminium). The not shadowed parts can be considered in vacuum. The light pipe connecting the scintillator to the photomultiplier is in Plexiglas. The number of ${}^6\text{Li}(n,\alpha){}^3\text{H}$ reactions in the scintillator is assumed to be proportional to the number of pulses registered by the scaler.

Figure 2. The central detector of the BSS. All dimensions are in cm.



The BSS studied in this work was exposed to ISO Reference neutron spectra, bare ${}^{252}\text{Cf}$ and ${}^{241}\text{Am-Be}$, at the ENEA Radiation Protection Institute (Bologna) using the shadow-cone technique (Figure 3) to correct for the room and air scattered neutrons. The results were used as experimental data for the proposed problem.

A preliminary validation study was necessary to check the reliability of the computational model against experimental data and literature independent data. The response functions calculated by Mares and Schraube, taken from literature [3] were folded on the ISO neutron spectra from ${}^{252}\text{Cf}$ and ${}^{241}\text{Am-Be}$. At the same time MCNP calculations of the responses of the full set of spheres were performed and measurements under irradiation from the same neutron sources were carried out. All the results were normalized to the 8” sphere values, obtaining a good agreement among the three independent sets of data (figure 4 and 5).

Figure 3. Calibration of LiI detector at the ENEA neutron ISO spectra facility (in figure the shadow cone is inserted).

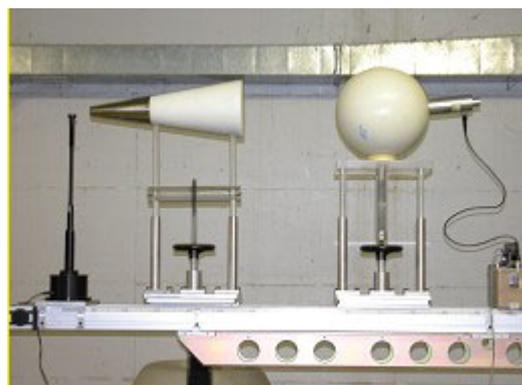


Figure 4. Comparison with measured, simulated data and data evaluated by Mares and Schraube for $^{241}\text{Am-Be}$ source.

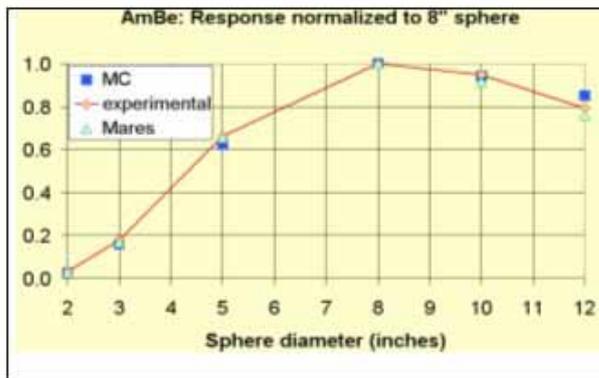
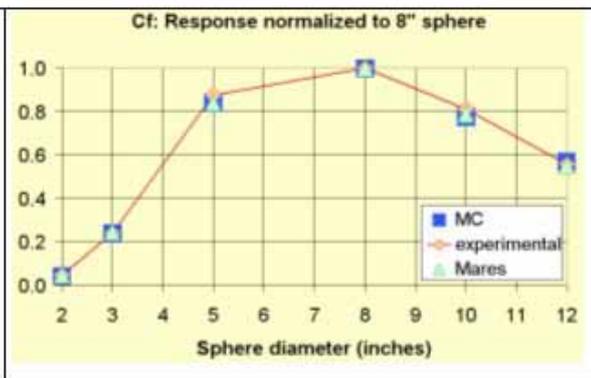


Figure 5. Comparison with measured, simulated data and data evaluated by Mares and Schraube for ^{252}Cf source.



2.3 Aim of the proposed problem tasks

Five tasks were proposed in the problem specifications. The first two tasks are related to the response of a minimal set of 4 spheres to the ISO reference sources. Participants are requested to determine the response of the 2'', 3'', 8'' and 10'' spheres exposed to a parallel neutron beam of the same cross sectional area of the sphere for the ^{252}Cf source: The detector axis should be parallel to the radiation field with the light pipe opposing the field direction. 5'', 8'', 10'' and 12'' spheres were used as a minimal set for the $^{241}\text{Am-Be}$ source, characterised by a higher mean energy. Mean energies are in fact about 2.13 MeV for ^{252}Cf and 4.16 MeV for $^{241}\text{Am-Be}$. The results were requested to be normalized to the 8'' sphere but also the absolute response values were required.

In the third task was asked to be investigated the response of the 8'' sphere, exposed to a parallel neutron beam of the same cross sectional area of the sphere, for a set of given energies and the result should be plotted normalized to the maximum value.

The fourth task consists in the study of the response of the complete set of the Bonner spheres.

In the fifth task the participant is required to determine the neutron spectrum within a inner thin lenticular layer (0.1 mm thick) placed inside the Bonner sphere towards the source and perpendicular to the beam axis in a given energy range. The evaluation of the neutron spectra for the two neutron sources entering in the detector is necessary for evaluation of the reliability of the transport and slowing down modelling within the polyethylene sphere.

To meet the minimal requirements of the problem, items 1) and 2) are highly recommended, since the algorithms to infer the neutron spectrum rely on the change of the detector response due to changes in the diameter or composition of the sphere.

Item 3) is a valid exercise to check the simulation model, since the required energy dependence can be easily compared with reliable literature data [3].

Item 4) is interesting in order to complete the study on the variation of the detector response as the moderating assembly changes, evaluating the overall consistency of the simulation model.

Despite the knowledge of the neutron spectrum at the centre of the sphere is not mandatory for a good management of the Bonner Spheres technique, item 5) helps to clearly understand the "spectrum shifter" action of the polyethylene moderator. The study of both $^{241}\text{Am-Be}$ and ^{252}Cf sources can clearly indicate why the response of a given sphere changes as the incident neutron spectrum changes. All irradiations should be considered in vacuum.

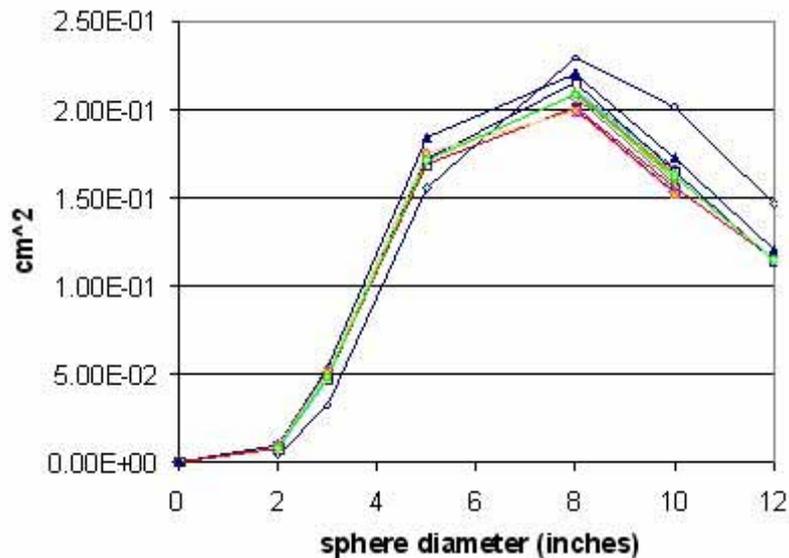
3. Discussion of the participant results

A significant number of 14 scientists took part in the solution of the problem, the largest majority using codes from the MCNP family. The first sub-problem obtained 12 replies with a reasonable agreement with the experimental data.

3.1. Items 1, 2 and 4

Items 1, 2 and 4 are concerned with the study of the response functions of spheres of various diameters under irradiation with ^{252}Cf and $^{241}\text{Am-Be}$. Figure 6 shows the absolute responses (counts cm^2) $^{241}\text{Am-Be}$ source. In general it should be noted that the response to the Cf source is higher than to Am-Be due to its softer spectrum which implies a more rapid thermalisation and moderation.

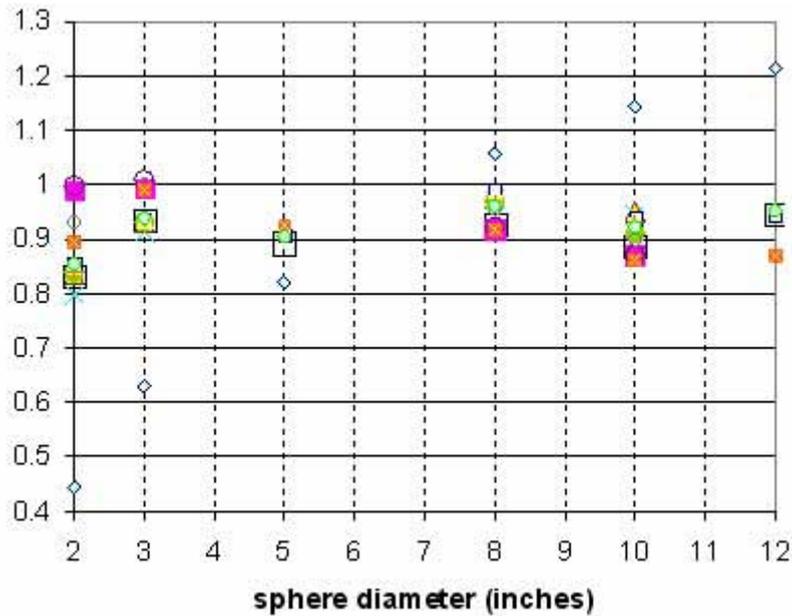
Figure 6. Absolute response as a function of sphere diameter for Am-Be



The two comparisons induce a series of consideration on the results:

- 1- the absolute counts per unit incident fluence are higher for the Mares and Schraube calculations. These systematic deviations can be ascribed to the different parameters characterizing both the crystal composition and geometry mock-up compared with the actual problem specifications.
- 2- Plotting the absolute response and not only the relative one is of high importance for a reliable analysis of the experiment, because a simple comparison of the data normalized (e.g.) on the 8'' sphere values can hide possible deviations present in the obtained results.
- 3- For the Cf source a general underestimation of the results compared with the experimental data should be noticed. This discrepancy is usually bounded within -10%, whilst some results differ of about -15%. In figure 7 data are presented. The agreement is quite good among participants with the exception of one of them that should be considered an outlier having provided results that cannot be interpreted (maybe related to some normalisation problem).
- 4- The results of Am-Be source are in a reasonably good agreement.
- 5- For Am-Be case the agreement with the experimental data is quite satisfactory, within +3% and -20% except for the 2'' diameter for which a large discrepancy should be noted that could be not explained.
- 6- In general it was noted that some of the participants did not provide a correct source spectrum to the code, in some cases shifting the source energy distribution of one group towards lower energies. This implied an erroneous softening of the source spectrum and an associated higher reaction rate even for low moderator thicknesses.

Figure 7. ^{252}Cf source: Ratios with the experimental data for the investigated sphere diameters.



3.2. Item 3

Item 3 is concerned with the study of the response functions of the 8'' sphere to monoenergetic neutron beams. The study was carried out for validation purposes. In figure 8 participants solutions are reported together with the Mares and Schraube and the authors of the problem solutions. In figure 9 the ratios with the authors' solutions are shown. The agreement is bounded within $\pm 8\%$.

Figure 8. 8'' Bonner sphere response to monoenergetic neutron beam.

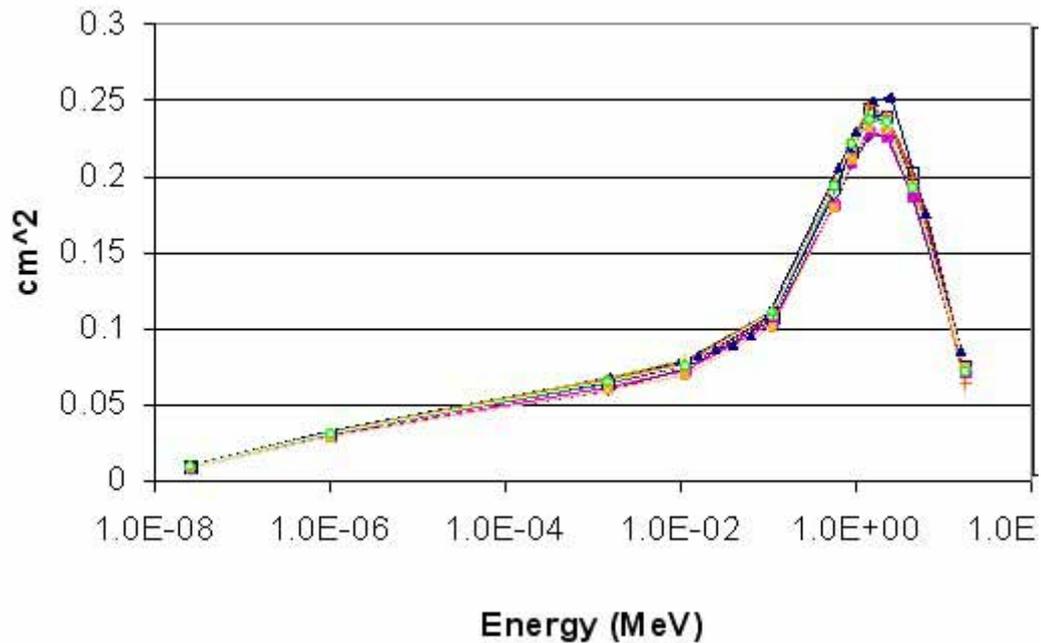
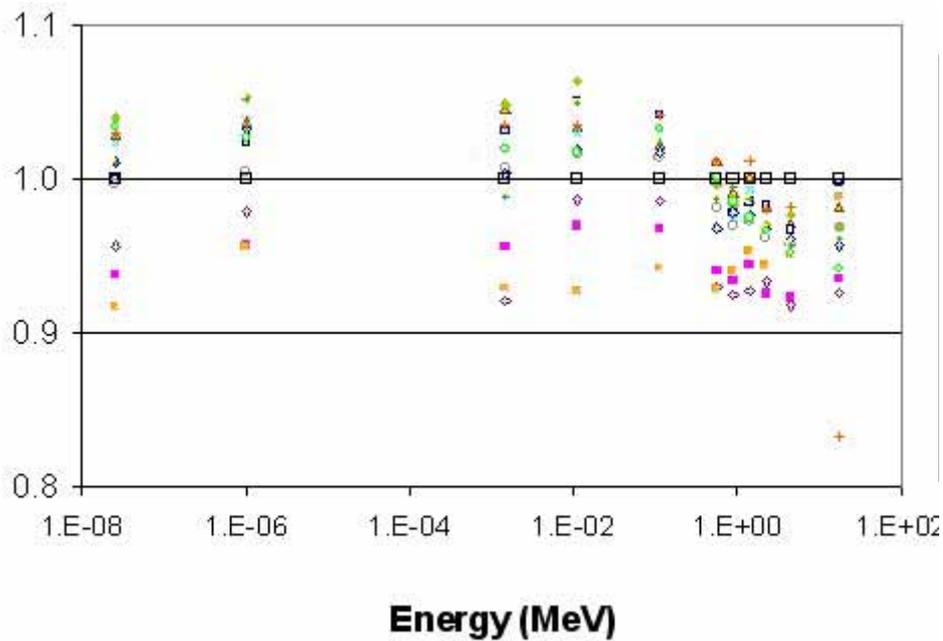


Figure 9. 8'' Bonner sphere participants response ratios



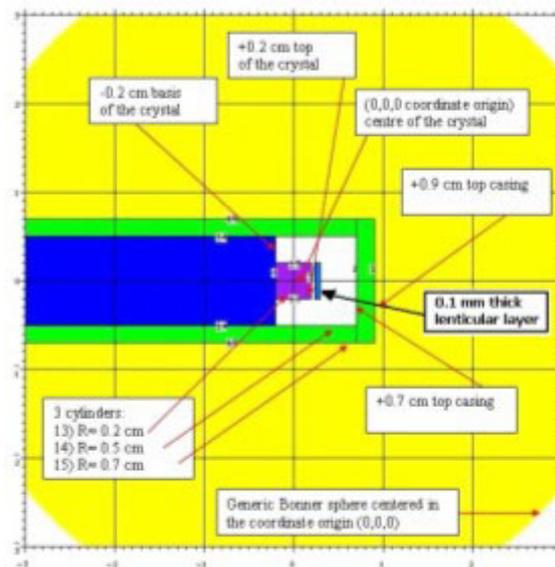
3.3. Item 5

In order to better analyze the consistency of the provided solutions it was of importance to evaluate the method the participants applied to treat the slowing down and thermalization of neutrons within the Bonner spheres. The spectrum change, during the transport, has a crucial influence on the reaction rate on the ${}^6\text{Li}$ nuclei that is responsible of the count rate of the instrument.

Therefore it was decided to compare the neutron spectra as calculated at the entrance of the LiI crystal, after crossing the 4 inches radius of the 8'' diameter sphere.

For this purpose an inner thin lenticular layer (0.1 mm thick) inside the Bonner sphere towards the source and perpendicular to the beam axis was ideally assumed and the spectral fluence estimator was applied inside this space region. In figure 10 the two-dimensional model of the geometry is presented.

Figure 10. 8'' Bonner sphere sketch with the inner lenticular layer for incoming neutron spectrum evaluation.



The participant's solutions have been evaluated according to the following scheme:

- 1- Normalization of the spectrum to the unit fluence: $\varphi_{\text{norm}} = \Phi(E) / \Phi$ where $\Phi = \int \Phi(E) dE$. Unit lethargy representation: $(E \varphi_{\text{norm}})$ vs. (E)
- 2- Graphical comparison with the "Reference" spectrum
- 3- Calculation of the following spectral indexes:
 - a) fraction of neutrons < 1 eV
 - b) fraction of neutrons within $[1 \text{ eV} - 10 \text{ keV}]$
 - c) fraction of neutrons $> 10 \text{ keV}$
- 4- Comparison of the spectral indexes with those obtained from the authors' solution.

Some preliminary considerations are necessary prior to the discussion of the results. It was assumed that all the participants were familiar with the physical meaning of neutron lethargy and with the expression of neutron spectra in lethargy units to represent the large energy range in which the neutron spectrum is usually spanned in a thermal fission source driven system (10 decades are to be represented and the areas of the plotted histograms should be representative of the neutron population within each energy group).

The subdivision into three spectral indexes is a classical system of reactor physics to provide a gross estimate of the spectrum behaviour.

With these premises it is now possible to point out that, despite the reasonably acceptable agreement on the integral quantities (see previous topics) a much worse agreement is obtained in the detailed neutron spectrum evaluation. In fact, both from the spectra plots and from the histograms of the spectral indexes it is evident that many participants did not evaluate their results (or probably were unable to correctly only express them).

For example the author's solution for Am-Be (see figure 11 and 12) provides 60% thermal, 10% epithermal and 30% fast fluence components. The participants solution range from 43 to 70% in the thermal component, from 5 to 20% in the epithermal and from 27 to 43% in the fast domain (apart from solution P2-D in which the thermal component is 3%, the epithermal 1% and the fast 96% probably due to an incorrect expression of fluence per unit lethargy).

The results spread was more limited for the Cf case (P2-D still with the same problem).

Figure 11. Neutron spectra solutions: comparison for Am-Be source.

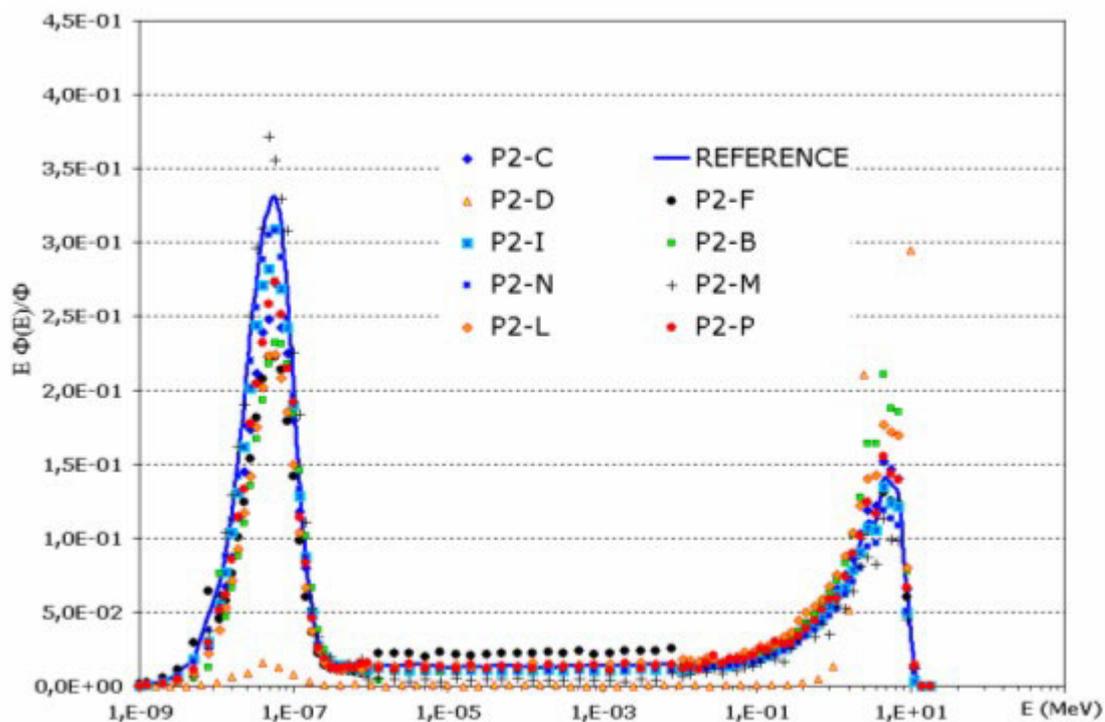
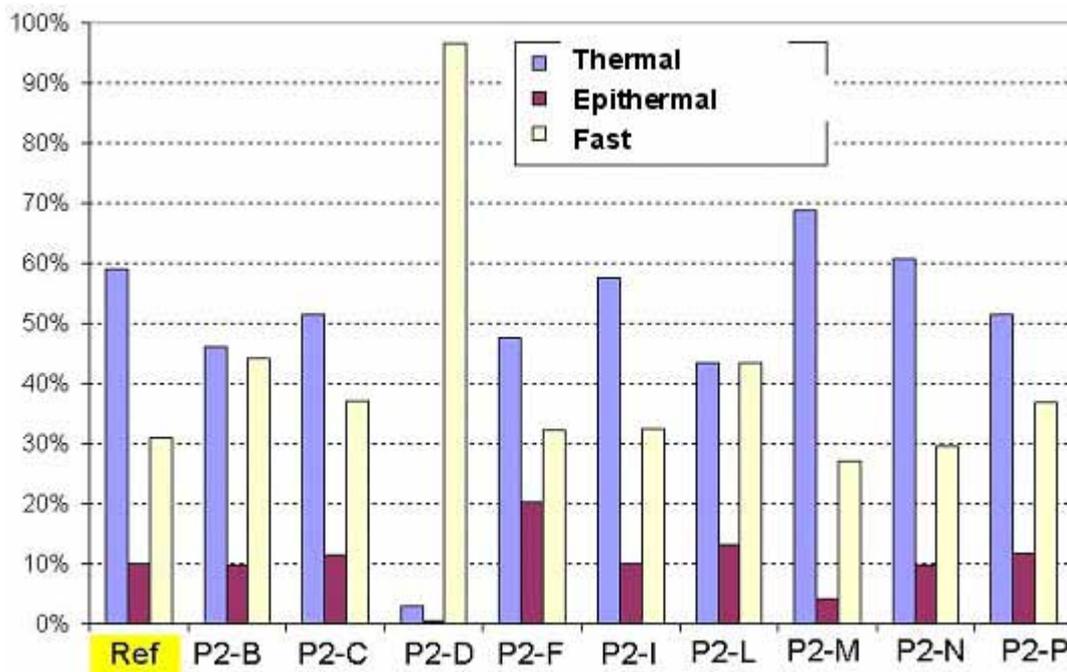


Figure 12. Spectral indexes comparison for Am-Be source.



4. Discussion of the participant results

The proposed problem, provided with experimental data obtained at a standard ISO neutron irradiation hall, had the scope to check the various methodologies applied to study the response of a widespread Bonner sphere spectrometer. Fourteen participants took part in the comparison also from outside Europe.

Despite the general rather satisfactory agreements that were obtained, still some important sources of deviations were detected. It was impossible to investigate in deep detail all the solutions due to the poor amount of information often supplied. Apart from some outliers that were present, the reasons of the discrepancies could be ascribed in general to:

- 1- Inadequate geometry and material composition modelling.
- 2- Inadequate expression of the source spectra (e.g. the determination of the mean energy of the source spectrum should be always employed as an indicator of the correct energy source sampling)
- 3- Incorrect choice of the reaction determining the counts rate. This is a crucial point of the modelling. In the specific case it had to be taken into account the ${}^6\text{Li}(n,\alpha)$ reaction.
- 4- Incorrect calculation of the number of atoms contained in the sensor volume (Nuclide atom density multiplied by the sensitive volume of the detector). This is necessary to calculate the absolute response per unit incident neutron fluence.
- 5- Wrong normalization to the unit incident fluence. This implies increasing the source disk area from which an expanded and aligned filed is impinging on a sphere of identical diameter. Therefore, for a correct normalization, the area of the source should be taken into account.
- 6- $S(\alpha,\beta)$ treatment for hydrogen bounded in polyethylene should be always taken into account.

All the proposed investigation topics, together with the performed measurements under standard ISO ${}^{252}\text{Cf}$ and ${}^{241}\text{Am-Be}$ sources had the scope to quality assure all the process connected to the determination of the response function of the employed set of Bonner sphere. The reliability of the response function plays a crucial role in the following unfolding procedure to be applied to obtain an unknown neutron spectrum in a given operative field. The uncertainties associated to the response evaluation propagate on the unfolded spectrum in a significant way. It is therefore necessary to

guarantee a suitable usage of the Monte Carlo modelling in this field together with appropriate validation against experiments.

On the other hand, on the basis of these results of the proposed exercise, it is clear how this kind of action should be considered as a valuable tool for preparing experts in this field and how a continuous education is fundamental also for researchers already involved in this applications.

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Avances en la puesta en marcha del proceso de gestión del conocimiento en la Autoridad Regulatoria Nuclear argentina

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AVANCES EN LA PUESTA EN MARCHA DEL PROCESO DE GESTIÓN DEL CONOCIMIENTO EN LA AUTORIDAD REGULATIVA NUCLEAR ARGENTINA

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RESUMEN

La gestión del conocimiento ha sido una innovación de gestión para Autoridad Regulatoria Nuclear (ARN) Argentina. Desde principios de 2006 se puso en práctica la gestión de los conocimientos críticos sobre la regulación de la actividad nuclear en el país que, el personal experto, pronto a jubilarse y ya jubilado, ha creado en décadas de investigación y aplicación. Argentina ha sido uno de los primeros países del mundo en desarrollar y fomentar la cultura de la protección radiológica. Un caudal importante de ese conocimiento científico técnico, ha sido dirigido hacia los usuarios de materiales radiactivos y responsables de instalaciones nucleares en el país y el exterior, en los cuales los expertos han participado durante décadas. Todo ese conocimiento e información debía ser gestionada para asegurarse que los actuales profesionales y las próximas generaciones tuvieran acceso a la misma en forma digital y organizada. También, se debía preservar la memoria de la institución que hace al relato histórico vigente. No se puede construir conocimiento sin un pasado registrado y sin referencias. Para fines del año 2006, se comenzaba a concretar una metodología que diera cuenta de esos importantes desafíos. Durante el año 2007 se avanzó en la implementación de distintas herramientas basadas en una metodología para la problemática. Esas herramientas teórico / prácticas incluyeron la puesta en práctica del programa “Historia del aprendizaje de los expertos de la ARN” basada en entrevistas en profundidad, y del proyecto “Preservación de la información y del conocimiento regulatorio histórico de la ARN”, basado en la digitalización de la información primaria de los expertos de la institución. La metodología utilizada para la gestión de esa información de los dos programas ha sido la elaboración de mapas de conocimiento. Esta herramienta práctica permite organizar, visualizar y distribuir fácilmente el conocimiento que ha sido explicitado y la información primaria de un experto o sobre un tema específico. Los mapas de conocimiento son una fotografía real de lo que se sabe sobre un tema en la institución, y esto permite saber qué planificar y sobre qué aspectos hay que hacer hincapié en la capacitación. Al mismo tiempo, los mapas de conocimiento han permitido crear la primera base de datos institucional de los expertos. Es este un avance significativo para el funcionamiento de una institución técnica que necesita de datos e información constantemente, como fórmulas para cálculos y datos históricos. Los programas y proyectos que se están llevando a cabo permiten visualizar de una manera clara las mejores prácticas que se utilizan en la organización pero que estaban implícitas en el personal. Indefectiblemente, el proceso de gestión del conocimiento está llevando a una situación de mejora continua en la ARN, ya que se están comenzando a aprovechar los conocimientos y la información de una manera más eficaz y eficiente. El tener la información y el conocimiento explícito en forma organizada y accesible permite ganar tiempos y costos en las búsquedas y hace más fácil la capacitación.

Mucha información que podía perderse con la jubilación de los expertos está siendo gestionada para que quede como reservorio y pueda ser utilizada por las futuras generaciones de reguladores nucleares. A modo de síntesis, el avance en la gestión del conocimiento en la ARN es importante en términos de efectividad del proceso, ya que con recursos limitados se ha logrado preservar y gestionar una cantidad de información que estaba destinada a perderse. Como todo nuevo modelo de gestión, el proceso tendrá sus tiempos para acomodarse como un proceso legitimado y necesitado por todas las áreas de la ARN. Creemos que a medida que vaya ocurriendo el cambio generacional y los nuevos profesionales demanden información que antes se obtenía de los expertos, las fuentes de información creadas por los programas serán de una utilidad relevante para la institución.

INTRODUCCIÓN

La puesta en marcha de la gestión del conocimiento en la ARN ha sido un proceso innovador para un organismo de la administración pública argentina, y esto está en relación con el proceso de modernización del Estado que se ha puesto en marcha desde la Secretaría de la Gestión Pública. Como se expresa en la memoria institucional del Proyecto de Modernización del Estado, “...nos propusimos instrumentar políticas tendientes a adaptar las capacidades estatales a un contexto político, económico y social cambiante...” (JGM: 2006)¹. En esa misma dirección, a la gestión del conocimiento la entendemos como una política institucional innovadora que mejora la capacidad estatal, en este caso de la ARN, de gestionar mejor uno de sus principales recursos: el conocimiento. Asimismo, el proyecto de gestión del conocimiento implementado en la ARN va en la dirección que se expresa el informe antes mencionado, en el sentido de que “...sostiene el desarrollo e introducción de herramientas modernas de administración que facilitan la gestión...” (JGM: 2006). A casi tres años de comenzar con la tarea, podemos evidenciar avances significativos en la forma de entender y encarar una problemática en el marco de la modernización de la institución. De esta manera la ARN coincide con el Proyecto de Modernización del Estado en lo que respecta a la institucionalización de nuevas herramientas para hacer más eficaz la acción del Estado.

En el caso concreto de la gestión del conocimiento implementada en la ARN, coincidimos con la definición tomada por la Administradora Gubernamental Graciela Falivene del Instituto Nacional de la Administración Pública Argentina (INAP) de Robert Logan (SIDEDEC: 2000, 23), el cual se refiere a que “...*La gestión del conocimiento es la actividad organizacional de crear un entorno social y unas infraestructuras técnicas, de tal forma que el conocimiento sea accesible, compartido y creado*”. Creemos en esa misma dirección en la necesidad de fortalecer una actividad organizacional que favorezca un ambiente humano y de infraestructuras de compartir el conocimiento para volver más eficaz a la institución. De varias formas, compartir el conocimiento moderniza la acción institucional en el sentido que genera espacios más democráticos de debate sobre algunas decisiones sin poner en riesgo la verticalidad y la autoridad en la toma de decisiones. Además, compartir el conocimiento fomenta la creación de más conocimiento en la institución. Si esta acción es llevada a cabo con responsabilidad y cuidado, la institución recibe *inputs* superiores, en términos de

¹ JGM: Jefatura de Gabinete de Ministros, Poder Ejecutivo Nacional, República Argentina.

información y conocimiento, para la toma de decisión. En ese sentido se crea nuevo conocimiento que puede ser aprovechado para volver más eficaz y eficiente a la institución.

En el caso de la ARN, luego de definir los conceptos fundamentales del proceso de gestión del conocimiento regulatorio, se pusieron en marcha dos proyectos que buscan generar una infraestructura técnica y humana para favorecer el conocimiento de la institución. Uno de ellos se denomina “Historia del aprendizaje de los expertos de la ARN”, que busca rescatar y explicitar información tácita e implícita sobre aspectos formativos del personal experto jubilado y pronto a jubilarse, para poderlos distribuir entre el personal como forma de complementar los conocimientos e información del personal, además de ayudar a la planificación de la formación futura de los cuadros. El segundo proyecto que se ha puesto en marcha se denomina “Preservación de la información y del conocimiento regulatorio histórico de la ARN” que tiene como objetivos centrales la preservación de la información y del conocimiento explícito de los expertos jubilados pronto a jubilarse y la gestión de esa información de una manera novedosa y segura. Ambos proyectos se complementan y generan estructuras técnicas y humanas nuevas para aportar a solucionar parte del problema focal de la institución, diagnosticado en 2006 por la Dirección de Planeamiento Estratégico y Reingeniería Organizacional (DIPRO) de la Secretaría de la Gestión Pública, como es la tendencia a la pérdida gradual de la capacidad regulatoria debido, entre varias razones, a la pérdida de los conocimientos técnico-científicos.

Los avances que se han logrado en la puesta en marcha del proceso de gestión del conocimiento han sido importantes ya que, de varias maneras y junto a las políticas de Calidad, se están generando nuevos espacios y un ambiente institucional de modernización de la ARN para intentar frenar esa tendencia de pérdida de la capacidad regulatoria, por lo menos en el aspecto de los conocimientos.

ASPECTOS HISTÓRICOS DEL PROBLEMA

La Autoridad Regulatoria Nuclear (ARN) actúa como entidad autárquica en la jurisdicción de la Presidencia de la Nación Argentina y fue creada el 2 de abril de 1997 por la Ley 24.804, denominada Ley Nacional de la Actividad Nuclear. La misma fue sancionada por el Poder Legislativo el 2 de abril de 1997. El Decreto N° 1390 del 27 de noviembre de 1998 reglamenta el accionar de la ARN.

Tres años antes de la creación definitiva de la ARN, en 1994, el Poder Ejecutivo Nacional separa, a través del decreto N° 1540/94, la actividad regulatoria del ámbito de la Comisión Nacional de Energía Atómica (CNEA), que manejaba esta actividad, creando el Ente Nacional Regulador Nuclear (ENREN). Al mismo tiempo separa, también de la CNEA, el manejo de las dos centrales nucleares de potencia en actividad –Atucha I y Embalse-, creando la empresa Nucleoeléctrica Argentina S.A. (NASA). En el caso del ENREN; la decisión se fundamentó, entre otras cuestiones, considerando que se debían reservar, como funciones propias del Estado Nacional la regulación y fiscalización de cada uno de los aspectos de la actividad nucleoelectrica, y asignando a una institución estatal independiente, el entonces creado ENREN, el ejercicio exclusivo de dichas funciones, a efectos de diferenciar el rol propio del controlante y del controlado. La separación de la CNEA de las actividades de producción de energía

nucleoeléctrica como de la regulación de todas las actividades nucleares implicó también el traspaso del personal necesario, y del conocimiento e información de los mismos, para el funcionamiento de las nuevas instituciones.

La Ley 24.804 define que la Autoridad Regulatoria Nuclear tiene como objetivo establecer, desarrollar y aplicar un régimen regulatorio para todas las actividades nucleares que se realicen en la República Argentina. Este régimen regulatorio contiene los siguientes propósitos 1- Proteger a las personas contra los efectos nocivos de las radiaciones ionizantes y mantener un grado razonable de seguridad radiológica y nuclear en las actividades nucleares desarrolladas en la República Argentina. 2- Asegurar que las actividades nucleares no sean desarrolladas con fines no autorizados por la ley y las normas que en su consecuencia se dicten, así como por los compromisos internacionales y las políticas de no proliferación nuclear asumidos por la República Argentina. 3- Prevenir la comisión de actos intencionales que puedan conducir a consecuencias radiológicas severas o al retiro no autorizado de materiales nucleares u otros materiales o equipos sujetos a regulación y control (ARN: 2006).

Antes del desprendimiento de la actividad regulatoria de la CNEA, varios de los fines establecidos por la ley nuclear eran llevados a cabo por la Gerencia de Protección Radiológica y Seguridad Nuclear de esa misma institución. Los logros alcanzados por esa Gerencia científica permitieron expandir y proyectar la protección radiológica y de la seguridad nuclear en nuestro país y en el mundo como un aspecto fundamental de la actividad nuclear. La masa crítica de conocimientos alcanzada por el personal de esa Gerencia tuvo fuertes impactos en las decisiones que se tomaron dentro de las actividades de la CNEA como así también en el ámbito internacional, como fue, por ejemplo, el gran aporte del experto argentino Dan Beninson en la publicación 60 del *International Commission on Radiological Protection (ICRP)*², que estableció una bisagra entre el pasado y el futuro acerca de los límites establecidos para las dosis de radiaciones ionizantes en el público y en los trabajadores de la actividad.

En términos de recursos humanos, la ARN recibió casi todo su personal experto

-profesional, técnico y administrativo- de dicha Comisión Nacional, trayendo consigo el *know how* acumulado durante todos los años de investigación y desarrollo regulatorio en la Gerencia de Protección Radiológica y Seguridad Nuclear.

Sin embargo, a partir las dudas y presiones de la Comunidad Internacional y del sistema político argentino sobre los fines del plan nuclear Argentino a principios de 1984, y con las consecuencias políticas en el mundo por accidente de la central nuclear de Chernobyl en la ex Unión Soviética en 1986, la actividad nuclear en Argentina comienza un proceso de desmantelamiento, que va a terminar plasmándose en la Ley 24.804 y la creación de las instituciones nombradas. Uno de los aspectos significativos de ese proceso fue el congelamiento paulatino primero, y total después, de los ingresos de personal para la renovación de los cuadros. Así, cuando en 1994 se crea el ENREN con el mismo personal de la CNEA, no se incorpora personal a la flamante institución. Tampoco cuando se crea la ARN en 1997.

Para ese año, el promedio de edad de los expertos rondaba los 55 años y conformaban una plantilla de aproximadamente 40 personas, sobre un total de 200 funcionarios. Pero para el año 2008, el promedio de edad de los expertos, ya ahora 25 sobre un total de 300

² www.icrp.org

funcionarios, llega aproximadamente a los 63 años. Esto significa que si bien ha ingresado un número alto de nuevos trabajadores, la cantidad de expertos ha disminuido, principalmente por jubilaciones. Este fenómeno ha generado lo que se denomina un “*gap*” generacional, que significa la ausencia en la institución de una masa significativa de trabajadores de 45 a 50 años de edad, que se ubicaría entre la generación de trabajadores de expertos que promedian ahora los 60 años y los nuevos trabajadores que promedian los 30 años de edad.

Este fenómeno que se ha generado en los últimos años evidentemente necesita de una solución práctica. La gestión del conocimiento si bien no tiene la capacidad de aportar soluciones concretas para el descongelamiento de los ingresos a la planta permanente, que brindaría un horizonte distinto a los nuevos trabajadores, bien puede aportar herramientas para acelerar la transferencia de conocimientos e información desde los expertos hacia el nuevo persona e intentar darle un freno a la pérdida de la capacidad regulatoria de la institución en el aspecto de los conocimientos.

PRESERVACIÓN DE LA INFORMACIÓN Y DEL CONOCIMIENTO

La preservación de la información y del conocimiento regulatorio histórico se presenta en la actualidad como una tarea fundamental para la ARN, porque la institución corre el riesgo de perder esa información debido a la jubilación de sus expertos en los próximos años; y en el caso que sea necesario recurrir a la misma para fundamentar decisiones y políticas en el futuro, existe el peligro de no poder acceder a la misma.

La información que tienen en sus oficinas y pc’s las personas expertas prontas a jubilarse ha permitido en el pasado poder fundamentar en forma científica muchas de las decisiones que se tomaron en la institución. El no acceso a las mismas deterioraría parte del contenido objetivo de las decisiones que se tomaran en un futuro. Al mismo tiempo, la no preservación de esa importante información degradaría la historia institucional, generando espacios vacíos para la formación de los nuevos trabajadores.

Una institución sin información ordenada y resguardada y sin conocimientos claros y transferibles de manera adecuada se puede ir convirtiendo con el paso del tiempo en una institución que podría basa sus decisiones en criterios de bajo nivel científico (sin basamento en la información y conocimientos objetivos de su personal) y orientándose a formas subjetivas de análisis y evaluación. En síntesis, los costos de perder toda esa información son realmente importantes, y en vista del relanzamiento del plan nuclear en Argentina, la ARN no puede permitir tal situación.

Como se busca no llegar a esos estados negativos en la institución, se ha puesto en funcionamiento el proyecto “Preservación de la información y del conocimiento regulatorio histórico”. El proyecto requirió al aval político del Directorio para planearse y ponerse en funcionamiento, porque se relevarían informaciones públicas pero también reservadas. Al mismo tiempo, se necesitó planificar y disponer de un presupuesto considerado, ya que mucha información debe ser tratada fuera de la institución, como son los planos de muchas de las instalaciones, y además se necesitó adquirir programas de software para poder gestionar toda la información rescatada.

La puesta en marcha de este proyecto requirió la participación de diversos actores dentro de la institución; fue así que se convocó a casi todas las áreas de trabajo de la ARN para que participaran en el mismo. La convocatoria fue exitosa y se pudo comenzar el rescate de la información con muy buena voluntad de varios expertos.

Metodológicamente, el proyecto se sustenta sobre la colaboración de las personas, más allá de una decisión tomada por el Directorio para llevar a cabo el proyecto. Sin la colaboración de los expertos para abrir sus oficinas y computadoras, el proyecto no podría ser llevado a cabo. También hay una investigación cualitativa sobre cada uno de los expertos, en tanto se los consulta e investiga sobre cuáles han sido sus tareas e informaciones críticas para la institución a través de entrevistas. Al mismo tiempo, se trabaja sobre con metodología documental que permite analizar la información seleccionada y poder así digitalizarla. El pilar fundamental de la metodología del proyecto es la digitalización documental de la información en papel relevada de cada experto.

La metodología fue consensuada con el sector técnico más interesado estaba en la puesta en marcha de un proyecto de estas características. Luego de un debate serio sobre cómo poder rescatar toda esa información, se sintetizaron las ideas en un proyecto y metodología posible.

Finalmente, para gestionar toda la información digitalizada y rescatada de los discos rígidos de los expertos se implementan mapas de conocimientos de cada una de las personas relevadas.

En términos de avances en la digitalización, durante los primeros meses del proyecto se han convertido a formato digital 190 páginas diarias, lo que equivale a 8 MB de información. Las horas/hombre empleadas para esto alcanzó en los primeros meses los 40 minutos por día. A medida que ha pasado el tiempo, el trabajo se ha hecho más eficiente y se ha llegado a la digitalización de 400 páginas por día, equivalente a 16 MB de información. Las horas/hombre empleadas rondo los 60 minutos por día.

Hasta julio de 2008 se han digitalizado más de 1000 documentos, correspondiente a más de 15.500 páginas y a 650 MB de información. El esfuerzo en horas/hombre de trabajo total para el escaneo ha sido aproximadamente de 78 horas.

MAPAS DE CONOCIMIENTO

Los mapas de conocimiento en una institución sobre un área específica de conocimiento o sobre los conocimientos de una persona es una herramienta interesante desde el punto de vista de la metodología de las ciencias sociales, porque permite analizar el mundo de informaciones y conocimientos que rodea a una persona o a un grupo de ellas. Con el mapa se pueden observar, ayudado por la construcción de un sociodrama de conocimientos, las relaciones sociales dentro y fuera de la institución que la el sujeto tiene y ha tenido en su vida. Al mismo tiempo, el mapa permite la localización exacta de la información relevante de la persona hacia una tarea. “Un mapa de conocimientos es un instrumento para proceder a la ubicación y codificación del conocimiento explícito y tácito existente en las organizaciones” (CIDEDEC: 2001).

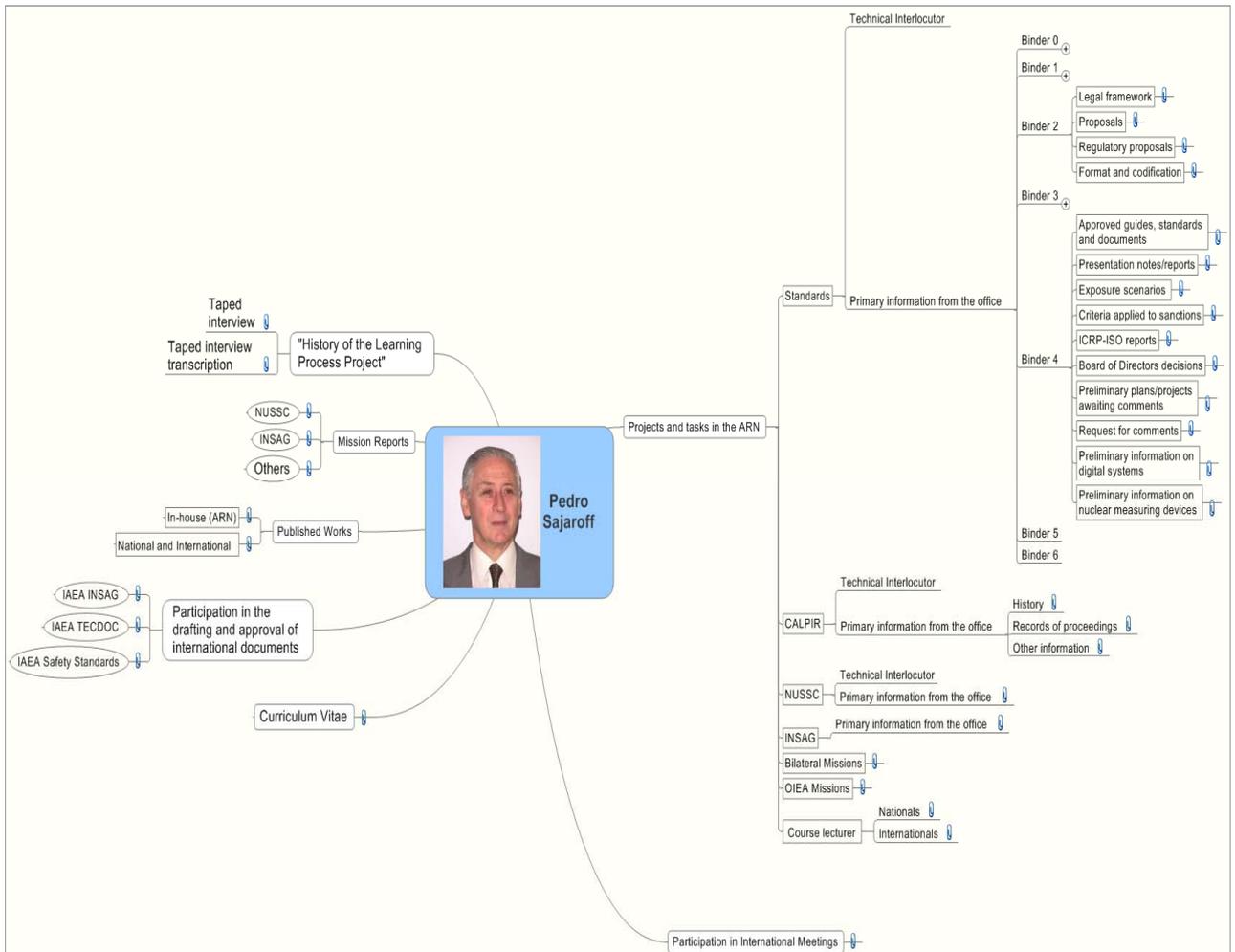
Es también una representación de conceptos, que puede agruparse en dominios de conocimientos. También es una herramienta que permite encontrar conocimientos y una navegación cognitiva. Se pueden realizar mapas de conocimientos explícitos, de implícitos o de conocimientos tácitos. De existir una elección importante de que tipo de conocimientos se quiere relevar en una organización, aunque en general, el conocimiento incluye todos esos tipos en forma integrada.

Una clasificación interesante sobre mapas de conocimientos es la que realiza la CEPAL (CEPAL: 2002), donde expone que las posibles técnicas de mapeo son: a) mapeo histórico o cronológico (muestra el estado del arte); b) mapeo bibliográfico (medición del contenido de publicaciones); c) mapeo cognoscitivo (conocimiento del individuo sobre un problema); d) mapeo conceptual (campo de conocimiento: conceptos, hipótesis, herramientas, modelos, teorías); y e) mapeos geográficos (distribución del conocimiento especializado en una región geográfica). Podemos agregarle una clasificación, en tanto que distinga entre mapas de conocimiento sobre información explícita y mapas de conocimiento sobre información implícita y tácita.

En el caso del proyecto de “Preservación de la información y del conocimiento regulatorio histórico de la ARN” se puso en marcha la construcción de mapas cognitivos y conceptuales, pero sobre los conocimientos explícitos; o sea toda la información en papel y digital que cada uno de los expertos ha construido y trabajado durante toda su vida en la actividad regulatoria. También, en varios casos, se implementaron mapas bibliográficos, teniendo en cuenta la cantidad de trabajos producidos por varios de los expertos.

Lo significativo de la construcción de estos mapas es que se pudo objetivar sobre qué sabe cada uno de las personas que se investigaron y de dónde obtuvieron ese conocimiento. Esto permite poder mostrarles y transmitirles a la nueva generación de trabajadores toda esa información de una manera organizada. En cada mapa de conocimiento de cada experto se pueden recorrer los temas de trabajo que ha desarrollado y sus ideas de cada uno de esos temas. Al mismo tiempo, el mapa actúa como una base de datos a los cuales se puede acceder en forma directa. Sin lugar a dudas que esta manera de presentar esta herramienta es una manera de democratizar el conocimiento y la información.

Como ejemplo presentamos un mapa de conocimiento de uno de los expertos.



CONCLUSIONES

La gestión del conocimiento, como una actividad organizacional y con una metodología sistemática, aporta al proceso de mejora continua que se ha puesto en marcha en muchas instituciones del Estado argentino a través de la Secretaría de la Gestión Pública. Esta nueva actividad organizacional permite el desarrollo de planes de acción que de varias maneras rompen con un modelo de administración pública estático. La gestión del conocimiento es en sí mismo un proceso de mejora continua, ya que pone en cuestión los preconceptos en la gestión de uno de los recursos más importantes de cualquier organización, el capital intelectual humano.

La metodología sistemática aplicada en el proyecto “Preservación de la información y del conocimiento regulatorio histórico de la ARN” nos ha permitido avanzar lenta pero eficazmente, ya que se están logrando los objetivos planteados al principio del proyecto. Esta mecánica de trabajo, planeada y sistemática, favorece la estabilización de la nueva actividad organizacional, la gestión del conocimiento. Y al mismo tiempo, genera en toda la institución un aporte al proceso de mejora continua que se busca institucionalmente, y que responde a las demandas sociales.

En síntesis, la continuidad de todo el proceso de gestión del conocimiento, que involucra distintos proyectos, recursos humanos específicos, aportes presupuestarios, entre otros aspectos, depende en gran medida de la capacidad de la institución de asimilar una nueva actividad organizacional dinámica y transversal. Así, la continuidad

de la modernización de la institución dependerá de la capacidad de la misma en generar actividades que fomenten la mejora continua como motor de la eficacia.

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Preservation of Primary Information Related to Radiological Protection and Nuclear Safety in the Argentine Nuclear Regulatory Authority

Chahab, M.

“Preservation of Primary Information Related to Radiological Protection and Nuclear Safety in the Argentine Nuclear Regulatory Authority”

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Abstract

The preservation of primary information related to Radiological Protection and Nuclear Safety in the Argentine Nuclear Regulatory Authority began as a need of and as significant contribution to the future activities of the institution. Since 2005 a high number of experts have retired from the organization and will continue to do so until 2010. Besides, the primary information that experts possess is technical information produced at the beginning of Argentina's regulatory activity in the 50's. If this information – on account of its relevance - could not be preserved properly or be made available to the future generation of scientists and technicians, such an issue could have a negative impact on the efficiency and effectiveness of the institution in the future.

The methodology selected for the project comprises several stages. Overall, the first stage consists in identifying primary information and expert's explicit knowledge through interviews and personal consultations. The second stage consists in converting to digital format the documentation that experts have traditionally kept in paper format. The third stage deals with transferring to a new database the already digitalized information from the computers of experts who are about to retire. The final stage is based on managing this information by creating knowledge maps and sociograms, experts personal Websites and a database with a megabrowser to make information readily accessible.

During the early months of the project, 190 pages have on average been converted to digital format on a daily basis, the equivalent of around 8MB of information. The men/hours employed for this task has been around 40 minutes per day. As time went by, the method turned more efficient and as a result, some 400 pages were converted to digital format on a daily basis, accounting for 16 MB of information. The men/hours employed for this task has been around 60 minutes per day.

Up until mid 2008, more than 1,000 documents have been turned to digital format, accounting for more than 15,500 pages and 650 MB of information. The effort in terms of total men/hours in scanning activities has been of around 78 hours.

The processing and management of information in digital format has been carried out through a software program that allows users to create knowledge maps and sociograms. This method, which is used to organize digital information, is a new management approach in the institution and will allow current and future Nuclear Regulatory Authority staff members to access information in a quicker and more comprehensive way.

One of the most immediate results has been the creation of a new database for the institution; another result is innovation in terms of information and knowledge management. Finally, the most important outcome has been the preservation of the primary and relevant institution-specific information that is the current and future basis of scientific and political decision-making at an institutional level. As an example of the results achieved, the knowledge map of one of the Argentine Nuclear Regulatory Authority experts is illustrated in this paper.

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Introduction

The preservation of primary information related to radiological protection and nuclear Safety in the Argentine Nuclear Regulatory Authority began as a need of and as significant contribution to the future activities of the institution. Since 2005 a high number of experts have retired from the organization and will continue to retire until 2010. Besides, the primary information that experts possess is technical information produced at the beginning of Argentina's regulatory activity in the 50's. If this information – on account of its relevance- could not be preserved properly or be made available to the future generation of scientists and technicians, such an issue could have a negative impact on the efficiency and effectiveness of the institution in the future.

Preserving this relevant information is not only related with the sizeable number of retiring experts but also with the “*Nuclear Renaissance*” in Argentina declared by the National Government in 2006. Announcements such as the completion of Atucha II nuclear power plant, the possible construction of a fourth nuclear power plant, the construction of reactor CAREN, and the start up of a uranium reprocessing facility impose on ARN new demands in terms of the regulatory actions to be taken, based on the scientific, technical and political knowledge of the institution for permit approval and operation regulation.

Many of the facilities announced by the National Government for the new nuclear plan had been planned in the past, with constructions beginning back in the 70's and 80's to be afterwards interrupted. All the regulatory information and expert knowledge of these facilities, which are currently being planned again, with constructions restarted, should be preserved and managed appropriately so that they can be used in the short- to mid-term by the new generation of officials. To this end, in late 2007, a specific process called “Preservation of Information and Historical Regulatory Knowledge in ARN” was started.

This effort is framed within the Regulatory Knowledge Management process that was also started in 2006, with a special focus on rendering explicit the tacit and implicit knowledge of retiring experts' learning process, based on a project called “History of the Learning Process”¹. As a result of such project - which is still ongoing - potentially relevant results are being achieved to plan and contribute to the training program for new and future staff members. Additionally to the project “History of the Learning Process”, the new project started in 2007 focuses on explicit information and knowledge. Considering that the only focus of the first project is on people and their scientific and technical education, the new one focuses on material objects, highlighting paperwork and files created by the experts.

In summary, this paper is an attempt to offer a brief sample of the objectives, methodology, tools, safety features and progress made by the project within the framework of ARN Regulatory Knowledge Management process.

Project Objectives

This project has been created to retrieve the primary information produced by the experts in ARN during the years they provided their services. Retrieval of such primary information involved converting to digital format the largest amount of critical information, entering it into an ad-hock database, and managing it with knowledge maps.

As a result of the project, all the material that was damaged was photocopied; all the new photocopies will be filed in the ARN library.

¹ The History of the Learning Process Methodology was presented by ARN in the International Conference on Knowledge Management in Nuclear Facilities. IAEA, 18-21 June 2007, Vienna, Austria.

The chief aim of the project is to retrieve critical information of retiring experts in paper format by converting it to digital format and entering it into the database. Retiring experts' digital information was also consolidated into the database.

Much of the information is related with scientific and technical knowledge that is the basis of many procedures and laboratory techniques, part of the regulatory day-to-day affairs. The political information underpinning political decisions made in the past by ARN have also been the focus of this project.

For the converted information we are using knowledge maps. The building of knowledge maps with the information gathered has been approached as another project objective, since this method allows accessing information in a simple and quick way, both at present and in the future.

For build knowledge maps needs adequate information and a previous know how. "A knowledge map is an instrument to conduct the location and codification explicit and tacit knowledge in organizations" (CIDEDEC: 2001). For good codification explicit and tacit information it requires a previous knowledge about how does it. Achieve this "know how" in a knowledge manager is a long process and we spend one year of work with that. Convert explicit tacit and implicit knowledge is a difficult process. However, with knowledge map tool this process is easier because we can see which knowledge and information are important and which does not.

Methodology

The project's methodology includes identifying, photocopying, and converting to digital format all experts' primary information. To this end, a thorough list of experts being 63 years of age or older was prepared for males and 58 years of age or older for females. A meeting was agreed with each expert asking them to cooperate with the project and help in identifying critical or relevant information for the institution. During the course of the meeting, a list of the tasks performed by them during their years of service in ARN is put together, and a list of the potential regulatory affairs they may have non-digital information of is also put together. The information surveyed is subsequently digitalized and a knowledge map per expert is created.

During the interview, experts are also asked who they have learned from each of the tasks they performed while on duty. This information is used to construct a knowledge sociogram for ARN.

Tools Used

The use of software tools has helped the project in arranging and managing retrieved information in a sound way. Software programs used in this project help scan documents, cut and paste information from PDF files, visualize information properly, create knowledge maps, transcribe interviews with experts, etc.

The contribution by ARN IT department has been truly critical in order to move forward with the project. The IT department has provided the necessary tools used at specific stages during the project. Some include creating direct dedicated connections between scanners and PCs to store the surveyed information, expanding hard drive memory to store information, and creating the relevant folders to arrange scanned and retrieved information from retiring experts' PCs.

Progress and Conclusions

The project on Preservation of Historical Regulatory Information and Knowledge has progressed slowly but it has nonetheless ensured that the retrieved information will be stored and managed in an appropriate and secure manner.

During the early months of the project, 190 pages have on average been converted to digital format on a daily basis, the equivalent of around 8MB of information. The men/hours employed for this task have been around 40 minutes per day. As time went by, the method turned more efficient and as a result, some 400 pages were converted to digital format on a daily basis, accounting for 16 MB of information. The men/hours employed for this task has been around 60 minutes per day. Up until mid 2008, more than 1,000 documents have been turned to digital format, accounting for more than 15,500 pages and 650 MB of information. The effort in terms of total men/hours in scanning activities has been of around 78 hours. Along this effort, a sizeable investment was made in order to digitalize drawings of facilities under past or present regulation by ARN. To this end, more than 350 drawings have been digitalized, accounting for 2.3 GB of information, with an equivalent effort in terms of men/hours for information management. It is expected that towards the end of 2008, some technical issues are solved to proceed to the scanning of 800 daily pages on average. Also, the project should have finished the digitalization stage of primary information of all the experts retiring in 2008. Knowledge maps will be finished and the relevant information will be available and readily accessible by all staff members in ARN.

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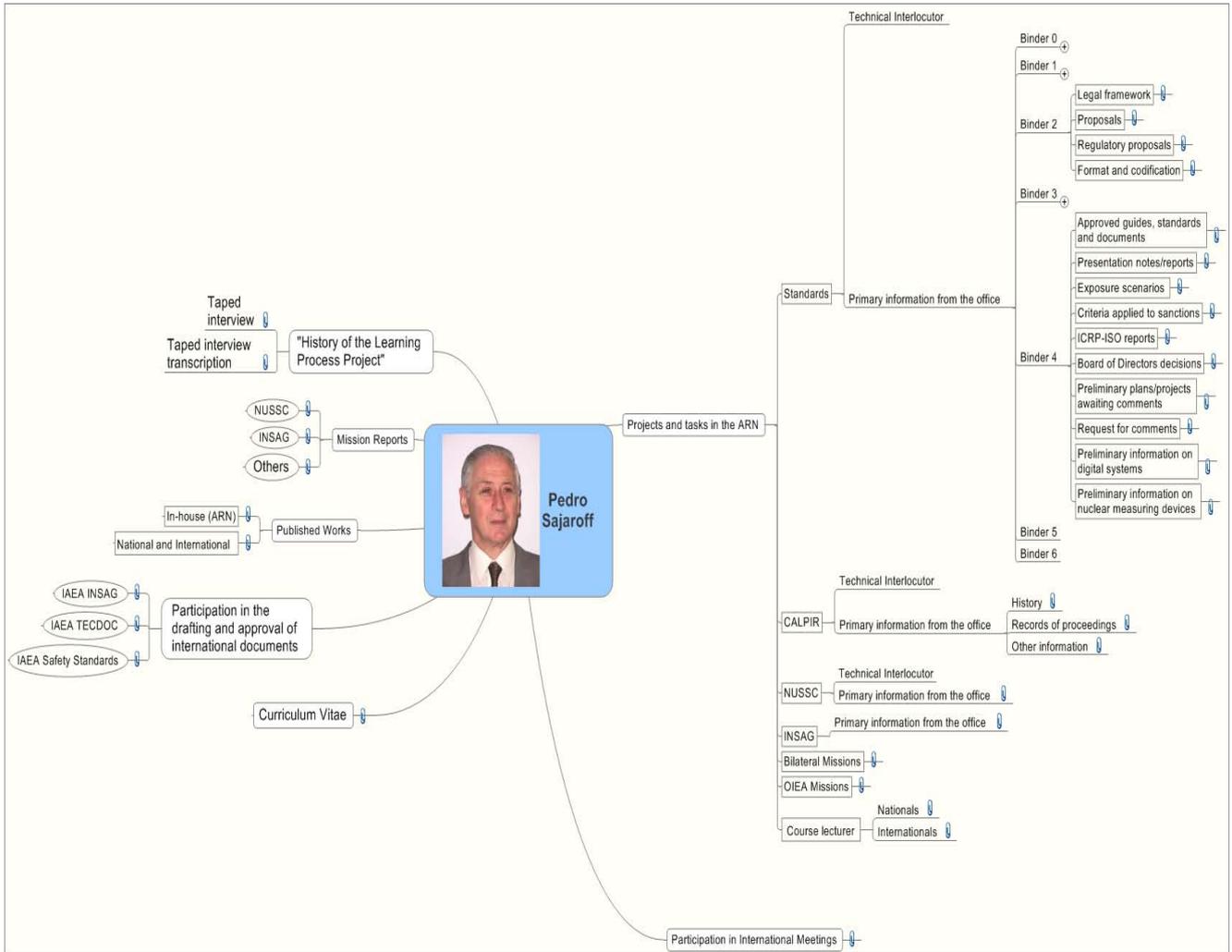
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Example of a Knowledge Map



Shielding of Medical Facilities. Shielding Design Considerations for PET-CT Facilities

Cruzate, J.A. and Discacciatti, A.P.

SHIELDING OF MEDICAL FACILITIES. SHIELDING DESIGN CONSIDERATIONS FOR PET-CT FACILITIES

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Abstract

The radiological evaluation of a Positron Emission Tomography (PET) facility consists of the assessment of the annual effective dose both to workers occupationally exposed, and to members of the public. This assessment takes into account the radionuclides involved, the facility features, the working procedures, the expected number of patients per year, and so on. The evaluation embraces the distributions of rooms, the thickness and physical material of walls, floors and ceilings. This work detail the methodology used for making the assessment of a PET facility design taking into account only radioprotection aspects. The assessment results must be compared to the design requirements established by national regulations in order to determine whether or not, the facility complies with those requirements, both for workers and for members of the public. The analysis presented is useful for both, facility designers and regulators. In addition, some guidelines for improving the shielding design and working procedures are presented in order to help facility designers' job.

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1. Introduction

1.1 Historical review

The first non-invasive technique to get images from the brain was a big revolution in itself. It was the radiography, invented in 1896, which used x-rays, discovered also by Wilhelm Röntgen. However, x-rays could show only the anatomical structures, and nothing else, i.e. it was an anatomical technique. The function of these structures could be inferred from anatomical changes, but only when they happened. In the early part of 1950, several suggestions were made in order to improve the quality of nuclear images for the detection of brain tumours and other brain diseases. In particular, it was suggested that the use of annihilation radiation following positron emission might improve the quality of brain images by increasing sensitivity and resolution. Finally, before the end of that year, the Physics Research Laboratory at Massachusetts General Hospital established a simple positron scanner using two opposed sodium iodide detectors. The results were encouraging. Data were obtained by translation of the two opposed detectors using coincidence detection with mechanical motion in two dimensions and a printing mechanism to form a two-dimensional image of the positron source. The success of the prototype positron scanner led to develop a scanner designed specifically for brain imaging^[1]. Figure 1 is a photograph of that first clinical positron imaging device at Massachusetts General Hospital (1953), and Figure 2 shows the corresponding scan of a patient with a recurring brain tumour. This PET scan resulted to be the first one in the world.

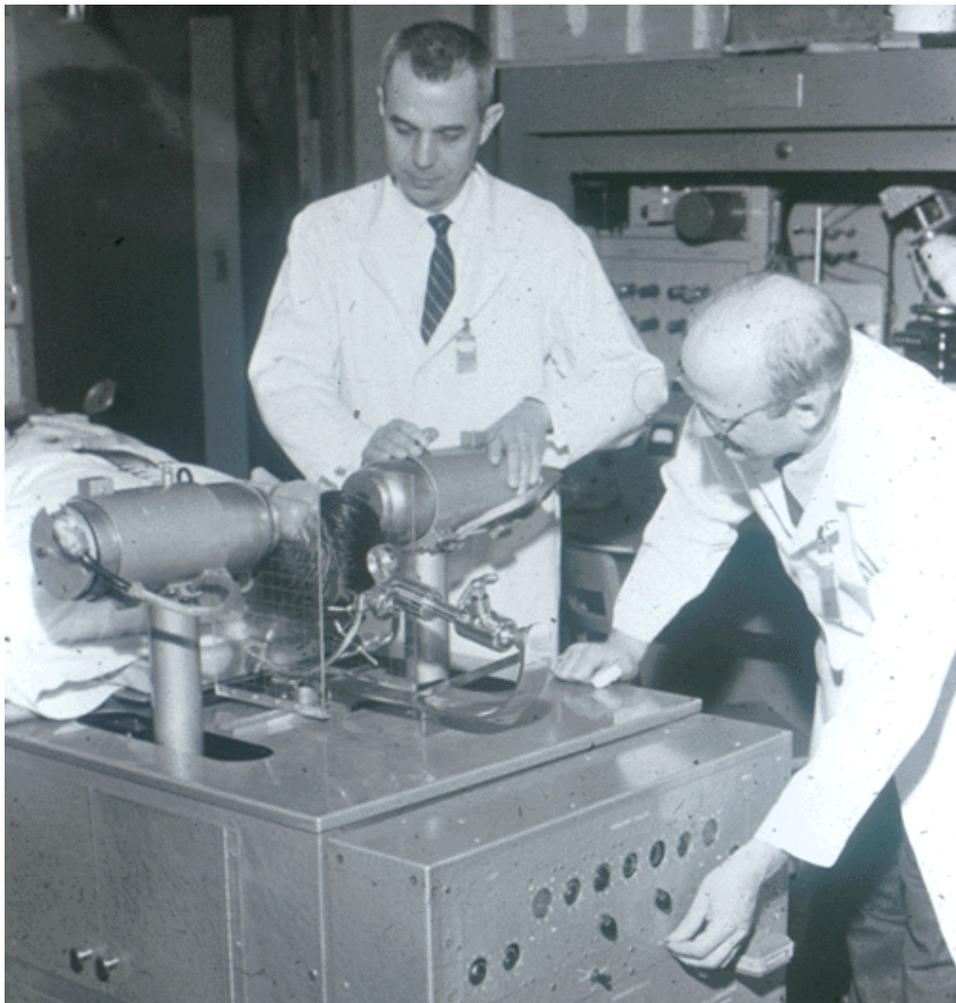


Figure 1. The first PET scanner in the Physics Research Laboratory at Massachusetts General Hospital (1953)

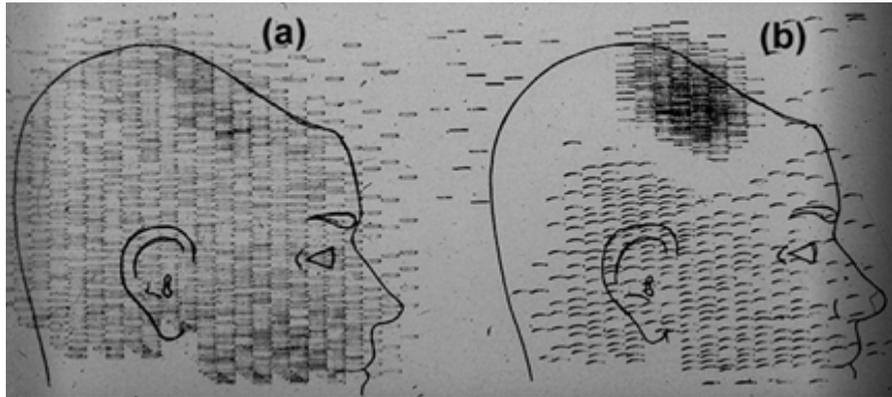


Figure 2. The first PET scan in the world (1953).

PET scanners developed parallel to technology during the following 50 years that came after the first development. Figure 3 shows the evolution of image quality from the first PET scanners to the latest and most sophisticated models^[2].

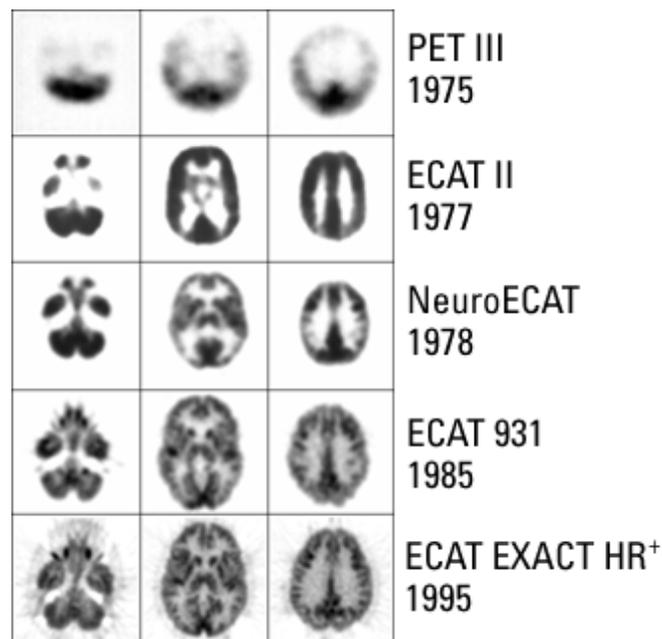


Figure 3. Evolution of image quality in PET scanners.

Modern PET scanners are very expensive sophisticated equipment. They are also much easier to install and to operate, with many new capabilities. PETs are so expensive that there are only about 150 installations around the world, which are mainly concentrated in USA, Europe and Japan. In the Southern Hemisphere, only Australia and Argentina have a very small number of PET facilities. Actually it's rather the whole facility, including the cyclotron for radiopharmaceuticals production, what push cost up^[2].

In some nuclear medicine centres, PET images are superimposed with computed tomography (CT) or magnetic resonance imaging (MRI) to produce special views, a practice

known as image fusion. These views allow the information from two different studies to be correlated and interpreted on one image, leading to more precise information and more accurate diagnoses (see Figure 4 for an example). In addition, manufacturers are now making PET/CT units, which are able to perform both imaging studies at the same time.

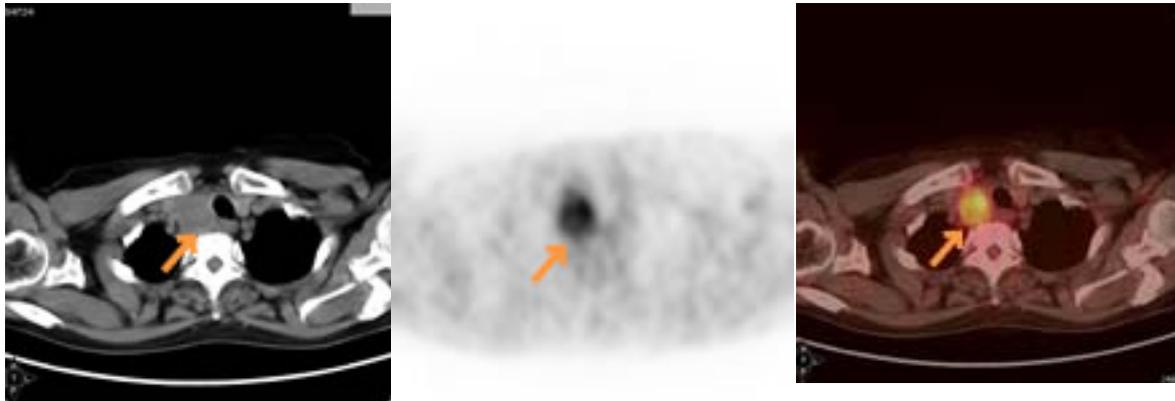


Figure 4. Example of image fusion at PET studies. Left: Detailed anatomy image from CT tomography. Centre: Image from PET study (no anatomy details present). Right: Image fusion of CT and PET images showing the tumour location into the detailed anatomy image.

1.2 PET at modern age

Positron emission tomography (PET) has been available worldwide for more than 23 years, but its use was not widespread until about 8 years ago. The power of PET resides in its ability to capture physiology and thereby obtain crucial diagnostic information unavailable from high-resolution pictures of the anatomy^[3]. This technique is a noninvasive diagnostic imaging tool that takes advantage of certain radiopharmaceuticals and allows abnormal metabolic activity in and around organs to be examined by injection of a radionuclide into a patient. As a result of the small chemical quantities administered to each patient, the radiopharmaceuticals do not disturb the physiologic processes of interest^[4].

Certain radionuclides decay by spontaneously converting a proton into a neutron and simultaneously emit an energetic positron (β plus decay). After the positron dissipates its kinetic energy as it traverses tissue (or other material), it captures an electron and forms a positronium atom. Because the electron and positron are antiparticles, they mutually annihilate, producing two 511 keV photons. The PET diagnostic technique is based on the detection of annihilation photons from a positron emitter radionuclide previously administered to a patient. Detectors positioned around the patient detect annihilation radiation resulting from the emitted positrons interacting with electrons. This annihilation radiation is released as two 511 keV photons that are emitted 180° apart. In the event that both of these photons are detected (gamma coincidence detection), the location of the annihilation interaction can be found using gamma detectors in a ring configuration. In Figure 5 can be seen a schematic of the complete detection system, while Figure 6 shows a real detector configuration used in commercial equipment. When the applications of PET and computed tomography (CT) are combined, a powerful diagnostic tool results: PET/CT. The PET portion provides molecular activity levels, while CT provides the anatomical structure of the body^[4]. The schematic for this system can be found in Figure 7.

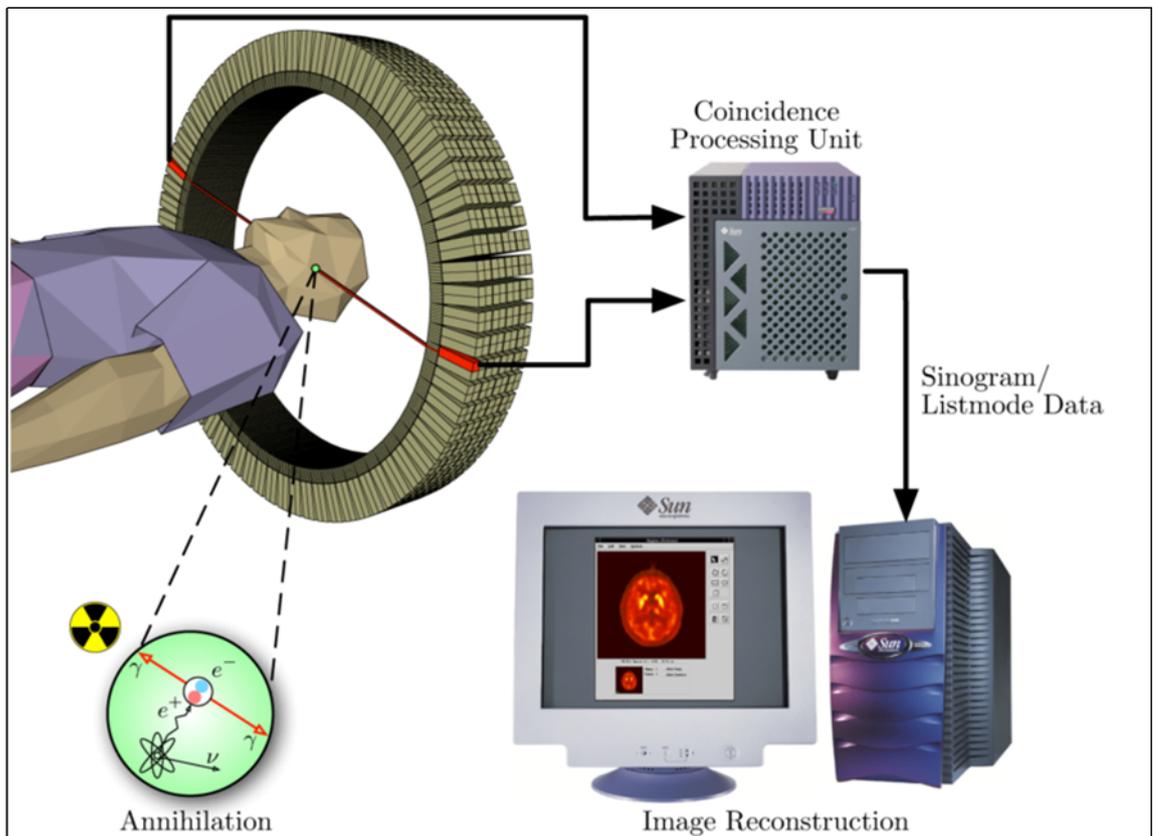


Figure 5. Schematic of the detection of scattered gammas in a PET scanner



A Siemens Ecat HR+ positron camera with front panel removed

Figure 6. Real detector configuration used in commercial equipment.

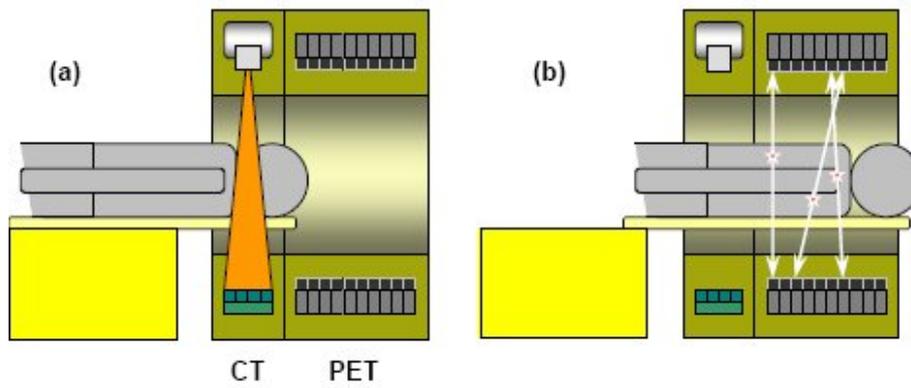


Figure7. Schematic of a PET/CT system

Figures 8a and 8b show a modern equipment which implements the PET/CT suite. The right picture in Figure 8b is the control room, with the modern computers that command the whole suite. The observation window is made of lead glass of thickness adequate for protection of operator from gamma radiation.

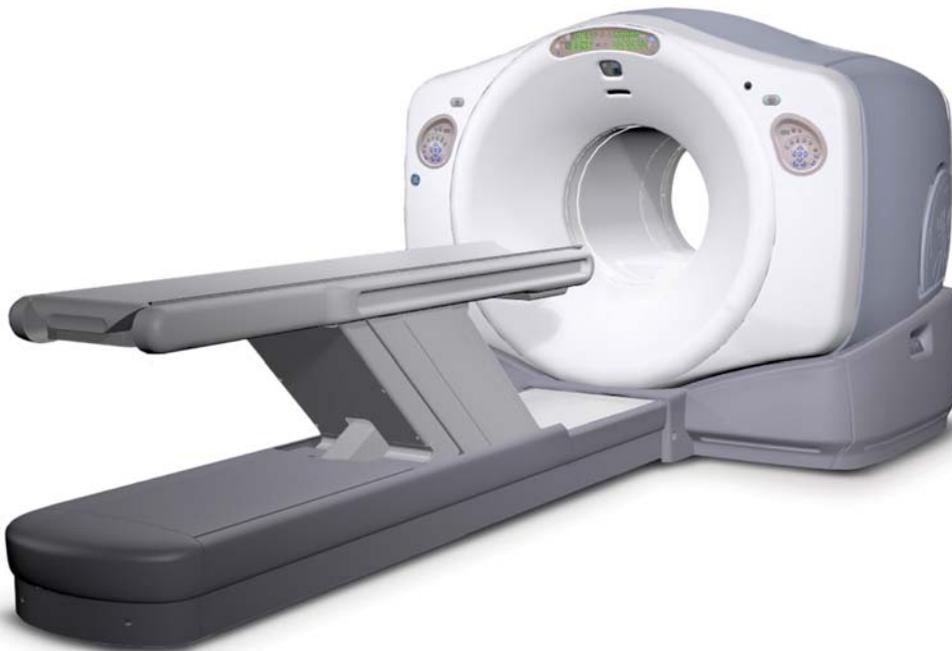


Figure 8a. Different views of a modern PET/CT suite

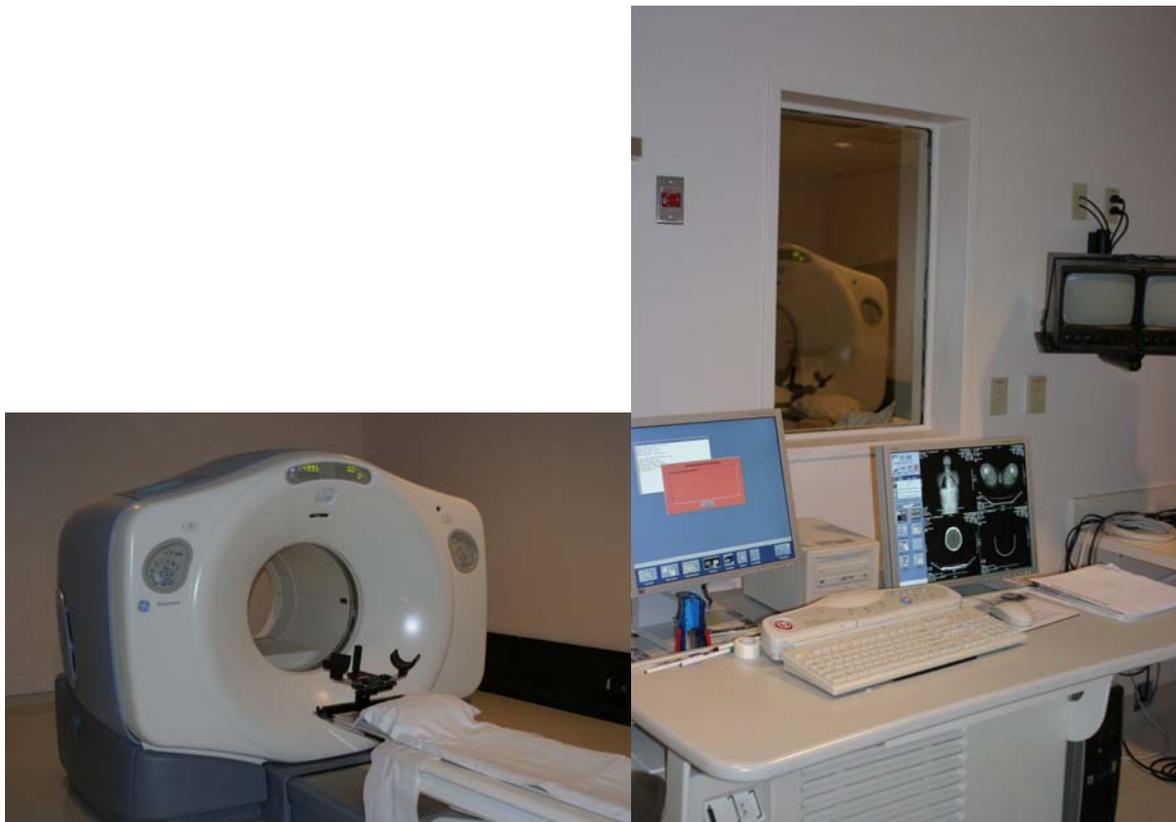


Figure 8b. Different views of a modern PET/CT suite

The recent explosion of interest in PET as a diagnostic imaging modality originates from mainly three factors: powerful radiotracers, coincidence detection technology, and study reimbursement^[3].

Being a diagnostic technique, the positron-emitting radionuclides more commonly used in PET (^{15}O , ^{13}N , ^{11}C and ^{18}F) have typically short half-lives and consequently many of them, such as ^{15}O , ^{13}N , and ^{11}C , have to be produced with an on-site cyclotron in order to have clinically useful quantities available (see Table 1 for more details). However, the 110 min half-life of ^{18}F is long enough to be regionally supplied.

The most versatile clinical PET radiopharmaceutical is F-18 Fluoro-2-deoxyglucose (F-18 FDG), a glucose analogue. F-18 FDG is accumulated in high concentration in metabolically active tumours as well as in the brain and the myocardium^[3]. A look at table 1 shows that ^{18}F has relatively longer half-life compared to the other commonly used positron-emitting radionuclides; on the other hand and foremost, ^{18}F is by far the most frequently radionuclide used in PET studies. Consequently, the expected radiation protection measures assuming that ^{18}F is always used should be adequate for studies where the same activity of shorter-lived radionuclides are applied. Because of this, this work focuses on PET centres that use ^{18}F .

Table 1: Dosimetric properties of commonly radionuclides used at PET.

Radionuclide	Energy [MeV]	Emissivity	Half-life
¹¹ C	0.511	2.00	20.4 min
¹³ N	0.511	2.00	10.0 min
¹⁵ O	0.511	2.00	2.0 min
¹⁸ F	0.511	1.93	109.8 min
⁶⁴ Cu	0.511 – 1.346	0.38 – 0.005	12.7 h
⁶⁸ Ga	0.511	1.84	68.3 min
⁸² Rb	0.511 – 0.776	1.90 – 0.13	76 s
¹²⁴ I	0.511 – 0.603 – 1.693	0.5 – 0.62 – 0.3	4.2 d

2. Evaluation procedure

In a typical practice, PET/CT patients are injected with an average activity of about 555 MBq (15 mCi) of F-18 FDG and instructed to lie down in what is called the uptake room for about 45-60 minutes while the radionuclide distributes throughout their body. They are then instructed to void their bladder of urine accumulation (in order to avoid signal interference due to gamma emission from bladder and also to reduce gamma dose in the bladder), and then taken to the scan room where they are given the approximately 20-30 minutes scan. By the time the patient leaves the PET/CT suite, the majority of the short-lived ¹⁸F (half-life is 110 minutes) has either voided from their body or physically decayed^[4]. Figure 9 shows a typical room layout used in nuclear medicine clinics. The shaded areas show the rooms specific to PET diagnostic: the hot lab (i.e. the place where all the radioactive sources are manipulated and stored until injection time), the uptake rooms (named “quiet” in this layout), the toilet exclusive to PET patients, and the scan room (PET/CT exam room in this case).

The radiological evaluation consists of the assessment of the annual effective dose to occupationally exposed workers and to members of the public. This evaluation takes into account the radionuclide involved, the characteristics and layout of the facilities, the working procedures and the expected number of patients per year. The evaluation embraces the distributions of rooms, the thickness and physical material of walls, floors and ceilings. The working procedures must give information of ¹⁸F activity administered per patient, the duration of time that each injected patient remains in the waiting room (uptake time), the duration of time of image acquisition (acquisition time), and so on. Occupational factors of the different areas are considered in the evaluation. Due to the characteristics of PET studies, the sources that are taken into account are: 1) radioactive sources in the hot lab, 2) the patient at rest during the uptake time, and 3) the patient in the PET scanner during the acquisition time.

Every PET facility must comply with the regulatory standards established in the country where the facility is located. These standards are mainly based on ICRP recommendations, the last being ICRP 103^[5]. These recommendations suggest the effective dose limits for planned exposure situations: 20 mSv per year for occupational exposure, and 1 mSv per year for public exposure. Similarly, the same recommendations suggest that every practice must be optimised: the likelihood of incurring exposures (where these are not certain to be received), the number of people exposed, and the magnitude of individual doses must be kept as low as reasonably achievable, taking economic and societal factors into account (ALARA principle). In order to be consistent with ALARA recommendations, shielding calculations use dose constraints instead of dose limits. These dose constraints are established by every national regulatory authority following the mentioned ICRP recommendations. Consequently, in order to verify the completion with regulatory standards, the annual effective dose results are compared with the dose constraints required by the corresponding national authority.

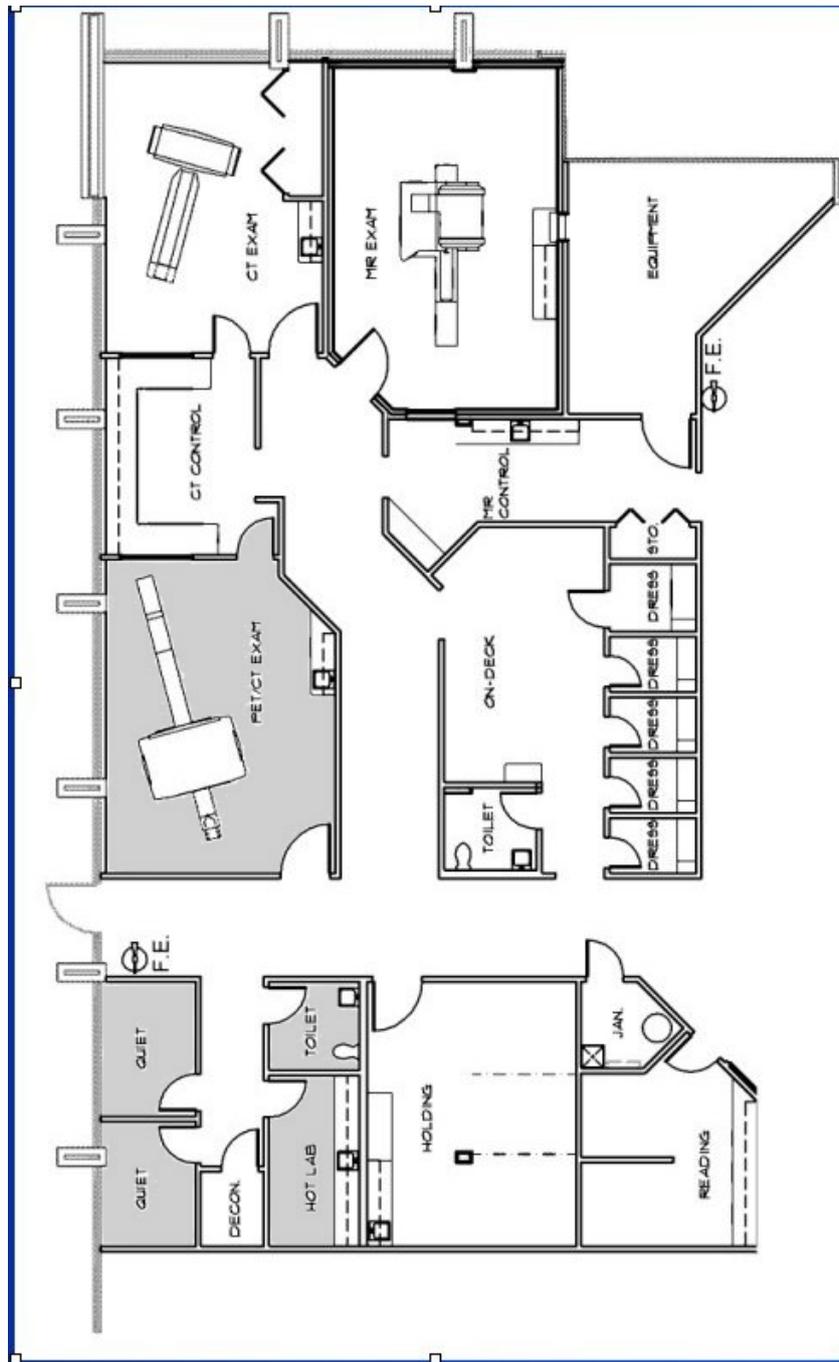


Figure 9. Room layout of a typical PET facility in a nuclear medicine clinic.

2.1 Parameters

Now, the parameters used in the evaluation procedure are discussed.

2.1.1 Effective dose rate constant, Γ

In radiation protection, the quantity of interest in shielding evaluation is effective dose; this quantity depends on incident photon energy and irradiation geometry. Figure 10 shows irradiation geometries encountered in common scenarios of radiation protection, and Figure 11 shows the effective dose dependence on specific irradiation geometries^[6]. During a given practice, it is usual to

have more than single irradiation geometry, consequently, in order to obtain the total effective dose one should use a combination of the mentioned geometries. Given an incident radiation, it is apparent from Figure 10 that antero-posterior irradiation geometry (A-P) yields the highest effective doses. Consequently, conservative assessment given by A-P geometry is used in shielding evaluation. Table 2 gives the corresponding values for this geometry.

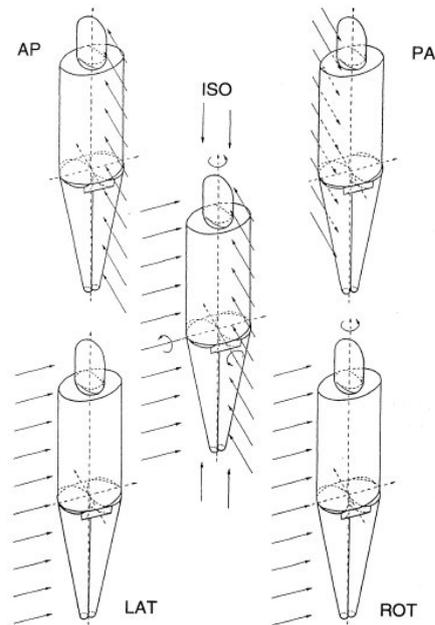


Figure 10. A Diagrammatic Representation of the Five Standard Irradiation Geometries selected by the ICRP and ICRU for the calculation of Conversion Coefficients in ICRU Report 57 (from ICRP Publication 74 and ICRU Report^{[6][7]})

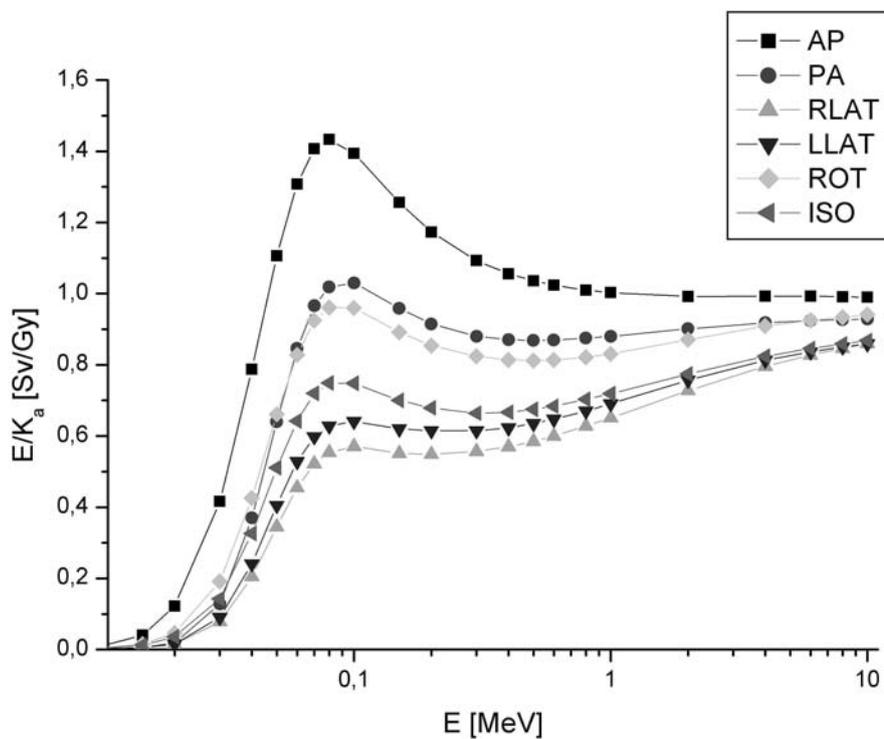


Figure 11. Effective dose per unit air kerma vs photon energy^[6]

Table 2: Photon effective dose conversion factor ^[6] in A-P geometry.

[MeV]	K_a/ϕ [pGy cm ²]	E/K_a [Sv/Gy]	E/ϕ [pSv cm ²]
0.010	7.43	0.00653	0.049
0.015	3.12	0.0402	0.125
0.020	1.68	0.122	0.205
0.030	0.721	0.416	0.300
0.040	0.429	0.788	0.338
0.050	0.323	1.106	0.357
0.060	0.289	1.308	0.378
0.070	0.298	1.407	0.419
0.080	0.307	1.433	0.440
0.100	0.371	1.394	0.517
0.150	0.599	1.256	0.752
0.200	0.856	1.173	1.00
0.300	1.38	1.093	1.51
0.400	1.89	1.056	2.00
0.500	2.38	1.036	2.47
0.600	2.84	1.024	2.91
0.800	3.69	1.010	3.73
1.000	4.47	1.003	4.48
1.500	6.14	0.998	6.12
2.000	7.55	0.992	7.49
3.000	9.96	0.993	9.89
4.000	12.1	0.993	12.0
5.000	14.1	0.993	14.0
6.000	16.1	0.993	16.0
8.000	20.1	0.991	19.9
10.000	24.0	0.990	23.8

Following these concepts, the effective dose rate constant of an unshielded point source is defined as the effective dose per unit time and per unit activity at 1m from the source. As stated before, in shielding evaluation antero-posterior geometry values are used. The effective dose rate constant for the radionuclides used in PET studies are calculated according to the conversion factors provided by ICRP Publication 74^[6] (see Table 2), using data from Table 1. The resulting constants may be seen in Table 3.

Table 3: Positron emitters radionuclides used in PET studies.
Half life ($T_{1/2}$), emissivity (ϵ), and effective dose rate constant (Γ).

Radionuclide	$T_{1/2}$	Photon energy [MeV]	ϵ	Γ [mSv m ² /h.MBq]
¹¹ C	20.4 min	0.511	2.00	1.44E-04
¹³ N	10.0 min	0.511	2.00	1.44E-04
¹⁵ O	2.0 min	0.511	2.00	1.44E-04
¹⁸ F	109.8 min	0.511	1.93	1.39E-04
⁶⁴ Cu	12.7 h	0.511 / 1.346	0.38 / 0.005	2.70E-05
⁶⁸ Ga	68.3 min	0.511	1.84	1.33E-04
⁸² Rb	76 s	0.511 / 0.776	1.90 / 0.13	1.50E-04
¹²⁴ I	4.2 d	0.511 / 0.603 / 1.693	0.5 / 0.62 / 0.3	1.45E-04

2.1.2 Administered Activity, A_0 / Uptake time, t_{up} / Acquisition time, t_{ac}

The amount of activity administered to a given patient depends on several factors such as mass of the patient, image acquisition mode, the length of the uptake time, PET tomograph, and so on. Generally adults receive 370-740 MBq (10-20 mCi) of ^{18}F while children receive 4-5 MBq / kg. As said above, after the injection patients lie down in the uptake room for about 45-60 minutes (uptake time) while the radionuclide distributes throughout their body. It is important for patients to be kept in a quiet resting state in order to reduce uptake in the skeletal muscles^[3]. After this, patients void their bladder of urine accumulation. Consequently, at the moment of the image acquisition the activity on the patient is

$$A(t_{up}) = A_0 \cdot e^{-\lambda \cdot t_{up}} - \alpha \cdot A_0 \quad (1)$$

where, $A(t_{up})$ is the remaining activity in the patient at the uptake time t_{up} , λ is the decay constant, A_0 is the administered activity and α is the fraction of the administered activity excreted by the patient before the acquisition of the images (typically $\alpha = 0.15-0.20$).

The information of the three mentioned parameters (the administered activity, the uptake time and the acquisition time) should be provided by the facility in the working procedure.

2.1.3 Transmission factor, k

The effective dose rate constant gives the dose rate produced by a given point source, with no shielding between the source and the point of interest. To take into account the effect of the shieldings interposed between any source and the point of calculation, a coefficient named transmission factor is used. The transmission factor is defined as

$$k = \frac{D(r_o)}{D_o(r_o)} \quad (2)$$

where, $D(r_o)$ is the dose at the interest point considering the shielding and $D_o(r_o)$ is the dose in the same point without considering the shielding. This factor may be defined for any source geometry, e.g. point or broad beam.

Since the body absorbs some of the annihilation radiation, the dose rate from the patient is reduced by a significant factor. Thus, in the evaluation of a PET facility, the building shieldings are considered as well as the patient attenuation (self-attenuation). Typical materials used as shielding in a PET facility are lead, iron, concrete, and masonry (common brick). The transmission factors applied were calculated by the Monte Carlo method^[3] considering broad beam geometry. In the original document, i.e. AAPM Task Group 108 : PET and PET/CT Shielding Requirements, transmission factors are given up to 50 mm for lead, 50 cm for concrete and 18 cm for iron. In this work, this range was extended by calculating the corresponding values with MCNPX Monte Carlo code^[8]. Table 4 shows the complete set of transmission factors for lead, concrete and iron, corresponding to the 511 KeV radiation. Figure 12 shows the plotting of the same transmission factors.

Table 4: Transmission factors for 511 KeV photon radiation

Thickness ^a	Lead	Concrete ^b	Iron
0	1.0000	1.0000	1.0000
1	0.8912	0.9583	0.7484
2	0.7873	0.9088	0.5325
3	0.6905	0.8519	0.3614
4	0.6021	0.7889	0.2353
5	0.5227	0.7218	0.1479
6	0.4522	0.6528	0.0905
7	0.3903	0.5842	0.0542
8	0.3362	0.5180	0.0319
9	0.2892	0.4558	0.0186
10	0.2485	0.3987	0.0107
12	0.1831	0.3008	0.0035
14	0.1347	0.2243	0.0011
16	0.0990	0.1662	0.0004
18	0.0728	0.1227	0.0001
20	0.0535	0.0904	2.50 10 ⁻⁵
25	0.0247	0.0419	7.88 10 ⁻⁷
30	0.0114	0.0194	
40	0.0024	0.0042	
50	0.0005	0.0009	
60	1.11 10 ⁻⁴	1.93 10 ⁻⁴	
70	2.38 10 ⁻⁵	4.14 10 ⁻⁵	
75	1.10 10 ⁻⁵	1.91 10 ⁻⁵	

^a Thickness in mm for lead, and cm for concrete and iron.

^b Concrete is considered to have a density of 2.35 g/cm³.

The patient self-attenuation depends on the energy distribution of the radiation source inside the patient itself. Studies found in the literature^[9] indicate that the mean transmission factor of a typical patient is approximately $k_{pat} = 0.64$ for radionuclides used in PET with a characteristic energy of 511 keV. For masonry, the transmission factors of concrete, modified by the relationship between densities, are used^[10].

$$\frac{\delta_{brick}}{\delta_{concr}} = \frac{X_{concr}}{X_{brick}} \quad (3)$$

where δ_{brick} is the masonry density, δ_{concr} is the concrete density, X_{brick} is the masonry thickness and X_{concr} is the equivalent concrete thickness. For shielding calculation purposes, masonry may be considered as a mix of common brick and mortar.

In case of a multilayer shield, the total transmission factor is obtained as a product of the individual transmission factors (included transmission of the patient, k_{pat})

$$k_{tot} = \prod k_i \quad (4)$$

where, k_{tot} is the total transmission factor for the whole shielding, and the k_i 's are the transmission factors of the individual ones.

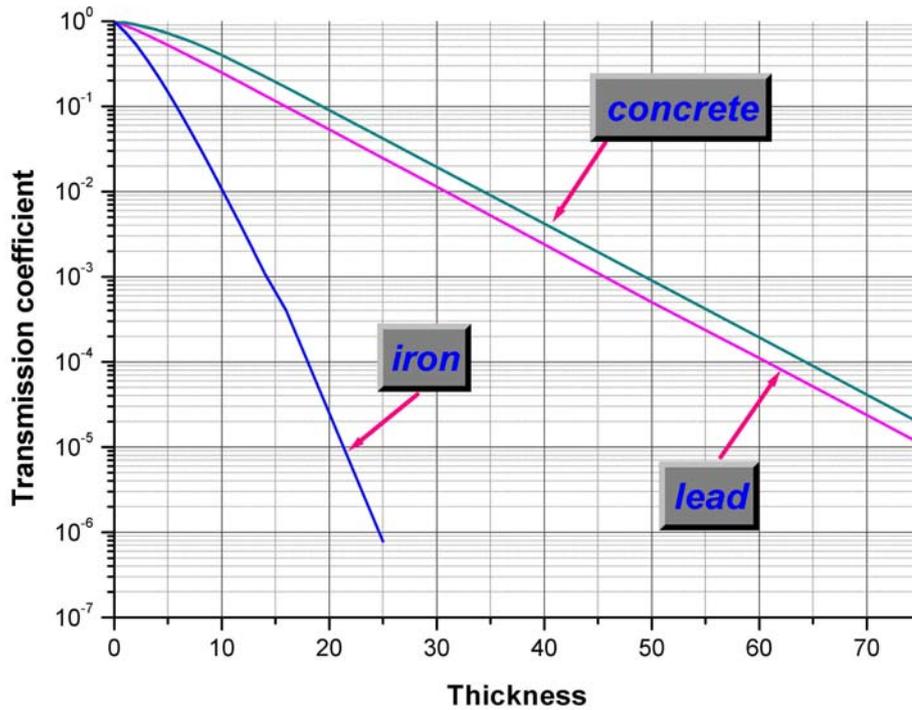


Figure 12. Transmission factors for 511 KeV photon radiation. Thickness in mm for lead, and cm for concrete and iron.

2.1.4 Dose reduction factor, R_t

As commonly radionuclides used in PET studies have short half-lives, it is necessary to consider their decay when the dose rate is integrated to yield the total dose. The result of the integration is

$$D(t) = \dot{D}_0 t \frac{1}{\lambda t} (1 - e^{-\lambda t}) \quad (5)$$

Looking at expression (5), we can define the so-called dose reduction factor as follows^[3]

$$R_t = \frac{D(t)}{\dot{D}_0 \cdot t} = \frac{1}{\lambda \cdot t} \cdot (1 - e^{-\lambda t}) \quad (6)$$

where, $D(t)$ is the effective dose accumulated during a time t , \dot{D}_0 is the initial effective dose rate and λ is the decay constant. In this way, the integrated dose is simply expressed as

$$D(t) = \dot{D}_0 \cdot t R_t \quad (7)$$

Table 5 shows dose reduction factor useful for typical situations encountered in PET diagnoses with ^{18}F . The value for 8 hours was supplied in order to represent a typical working day.

Table 5. Typical values for the dose reduction factor

t	R _t
30 min.	0.911
45 min	0.871
60 min.	0.832
90 min.	0.762
120 min	0.701
8 hours	0.314

2.1.5 Occupancy factor, T

The occupancy factor (T) for a given area or point of interest is defined as the average fraction of the effective irradiation time that a maximally exposed individual remains in the area or point of interest. It is important to stress that the occupancy factor for an area is not the fraction of time that it is occupied by any persons, but rather it is the fraction of time that it is occupied by the single person who spends the most time there. Thus, for example, a waiting room might be occupied at all the times during the working day, but have a very low occupancy factor since no single person is likely to spend more than 50 hours per year in a given waiting room. This value would imply an occupancy factor of $T = 50/2000 = 1/40$ for a normal working year. Instead, conservatively $T = 1/20$ is suggested by NCRP 151 (see Table 6). Occupancy factors in uncontrolled areas will rarely be determined by visitors to the facility or its surrounding areas, who might be there only for a small fraction of the year. The maximally exposed individual will normally be an employee of the facility^[10].

A value of unity is usually assigned to the occupancy factor for controlled areas. However, there can be situations in which access to a controlled area is restricted even for radiation workers when radiation is being produced (e.g., the control room). In such cases, the qualified expert designing the shielding requirements for that controlled area may use local information on occupancy of the area^[10].

In conclusion, the occupancy factor depends on both, the facility itself and the work procedure. In those cases where this information is not available, it is possible to apply the T values suggested in the bibliography^[10]. Table 6 shows these suggested values.

Table 6. Suggested occupancy factors (for use as a guide in planning shielding when other sources of occupancy data are not available)^[10].

Location	Occupancy Factor T
Full occupancy areas, administrative offices, treatment planning areas, treatment control rooms, nurse stations, attended waiting rooms, occupied space in nearby building	1
Adjacent treatment room, patient examination room adjacent to shielded vault.	1/2
Corridors, employee lounges, staff rest rooms.	1/5
Treatment vault door.	1/8
Public toilets, unattended vending rooms, storage areas, outdoor areas with seating, unattended waiting rooms, patient holding areas, attics, janitors' closets.	1/20
Outdoor areas with only transient pedestrian or vehicular traffic, unattended parking lots, vehicular drop off areas (unattended), stairways, unattended elevators.	1/40

2.2 Calculation model

As mentioned above, basically two types of radioactive sources are present in the facility: the patients themselves and the sources located in the hot lab. The sources in the hot lab are most of the time stored into a lead container, typically 5cm-thick. In Figure 13 can be found a typical lead housing into the hot lab with a raising device for manipulation of the shielded vial package. According to Table 4 the transmission factor in that case is $5 \cdot 10^{-4}$. As a consequence, doses due to these sources are usually negligible. In this work, only doses due to patients are considered; however it is recommended to do the dose assessment for each particular case.



Figure 13. Typical shielding for temporary storing of PET sources.

In all the calculations, a simplified model is used by considering point sources. This model is conservative enough to be used in radiation protection. As stated before, the quantity of interest in radiation protection is effective dose, which normally is expressed in annual form. Using formula 7 for the case of a point source, the annual effective dose at a point of interest, produced during a specific operation with that point source is given by

$$D = \Gamma \frac{A t_{op} N}{d^2} k R_t T \quad (8)$$

where, Γ is the effective dose rate constant of the source, A is the initial activity present into the patient at the beginning of the operation, t_{op} is the time length of the operation, N is the number of times that each operation is performed annually, d is the distance between the source and the point of interest, k is the total transmission factor taking into account all shields between the source and the point of interest (including the patient), R_t is the dose reduction factor and T is the occupancy factor at the point of interest. The only operations considered here are: 1) the patient at rest (uptake time) and 2) the patient in PET scanner (acquisition time). The total annual effective dose is calculated by adding the doses produced in both situations.

Because the 511 keV annihilation photons are so penetrating, it is necessary to consider controlled and uncontrolled areas above and below the PET facility as well as those adjacent on the same level. These also include all surrounding uncontrolled areas in the vicinity of the PET imaging clinic. Figure 14 shows generally accepted source and target distances that apply in these cases. Typically, one assumes that the patient (source of the activity) is 1 m above the floor. The dose rate is

calculated at 0.5 m above the floor for rooms above the source, and at 1.7 m above the floor for rooms below the source^[10].

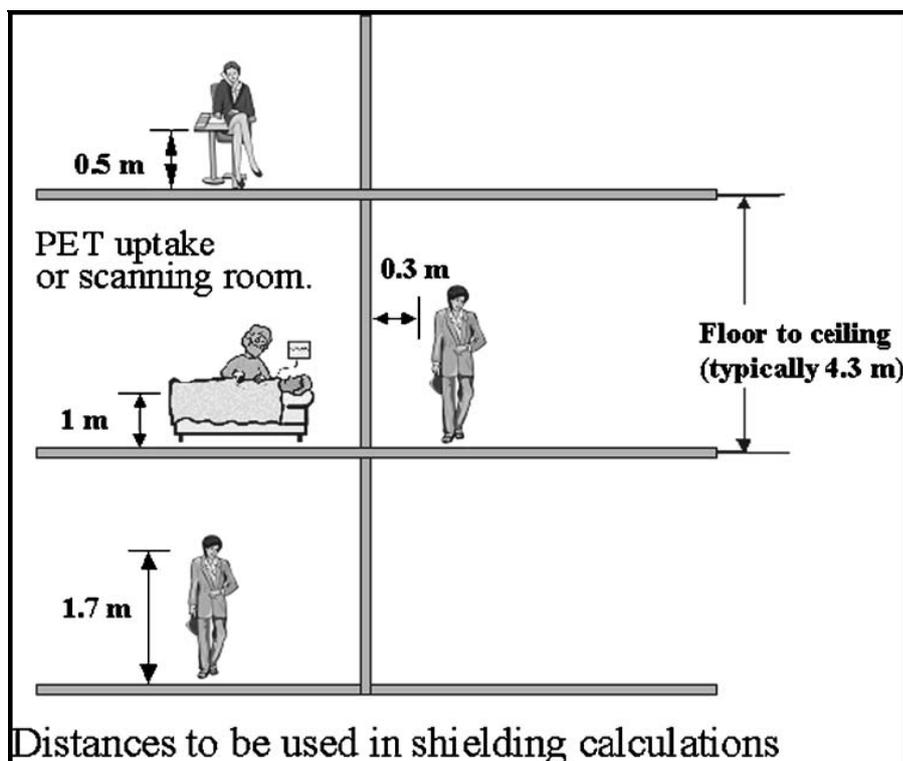


Figure 14. Generally accepted distances for PET calculations.

3. Evaluation guidelines and design considerations

In this section some evaluation guidelines are presented. Some simple cases are analysed applying the previous formulae, in order to draw conclusions about radiation protection. On the basis of the outcome a series of suggestions are made, which intend to provide a guide to improve both working procedures, and the facility design.

It has to be stressed that radiation safety must be guaranteed by the facility design itself and not by particular procedures. For example, if access to a given controlled area into the facility may not be restricted either by personnel or by any physical means, then shielding must be adequate for protection to the public, considering an occupancy factor of 1. This concept is of capital importance when considering surrounding uncontrolled areas in the vicinity of the PET facility. In this case, an occupancy factor of 1 must be used, because the facility itself cannot ensure the use given to that area: may be a garage nowadays, but a private building in the future. Instead, an uncontrolled area within the facility may be different because, in this case, it could be possible that facility can control the use given to that area.

Now, some guidelines for improving the shielding design of a typical PET facility will be presented. An average administered activity of 555 MBq (15 mCi) of ^{18}F , an uptake time of 60 min, and 40 patients per week will be supposed. For the sake of illustration, dose constraints required by the Nuclear Regulatory Authority of Argentina will be used: 5 mSv per year for occupational exposure, and 100 μSv per year for public exposure^[11].

In a complete PET study, a set of operations is performed (detailed in the working procedure of the clinic), and each operation is performed one time per each patient. In the typical clinic described

here 40 patients are attended per week, so that, the number of times each operation is repeated during one year would be equal to the number of patients per year, i.e.

$$N = 40 \frac{\text{pat}}{\text{week}} \times 52 \frac{\text{week}}{\text{year}} = 2080 \frac{\text{pat}}{\text{year}}$$

Prior to injection to patient, the needed radioactive aliquot has to be transferred from the vial to the syringe. This operation is carried out interposing a lead shield between the vial and the operator. The shield used in this case is the lead glass that is structural part of the top cover of the lead container (see Figure 13). This operation may be seen at Figure 15.



Figure 15. Transferring of the radioactive aliquot from the vial to the syringe.

Because of the high effective dose constant associated with positron-emitting radionuclides, hand doses for individuals drawing up and administering PET radiopharmaceuticals can be substantial. For example, the dose rate at 5 cm from an unshielded syringe with a typical administered activity is

$$\dot{D} = \frac{\Gamma A}{d^2} = \frac{1.39 \cdot 10^{-4} \text{ mSv m}^2 / \text{h MBq} \times 555 \text{ MBq}}{(0.05 \text{ m})^2} = 30.86 \text{ mSv/h}$$

that is, about 31 mSv/h. Assuming, conservatively, that each application takes 1 minute, the annual hand dose to a technician would be

$$D = 31 \frac{\text{mSv}}{\text{h}} \times \frac{1}{60} \frac{\text{h}}{\text{pat}} \times 2080 \frac{\text{pat}}{\text{year}} = 1070 \frac{\text{mSv}}{\text{year}} !!!$$

Tungsten syringe shields can reduce the hand dose, but the additional weight can make injections difficult. An example of a syringe shield may be seen at Figure 16.



Figure 16. Typical syringe shield used in PET studies.

On the other hand, after the syringe was loaded with the radioactive solution, it has to be translated through the facility up to the uptake room, where the patient is injected. In order to minimize dose to technicians (and to a lesser extent to members of the public), this translation is made with the syringe located within a cylindrical lead shielding (see Figure 17).



Figure 17. Cylindrical lead shielding for syringe translation.

The injection may be administered to the patient using a catheter, with the syringe located within the mentioned lead shielding (see Figure 18), but other ways to reduce hand dose are to use automatic dispensing systems and to divide the injection responsibilities among the staff.



Figure 18. Administration of an injection to the patient.

If a single member of the staff would remain at 1 m from the patient during uptake time, then the dose to this technician would be (according to formula (8))

$$k = 0.64 \quad (\text{patient attenuation})$$

$$t_{op} = 1 \text{ h} \Rightarrow R_t(1\text{h}) = 0.832 \quad (\text{dose reduction factor for one hour integration})$$

$$T = 1 \quad (\text{full occupation})$$

$$D = \frac{1.39 \cdot 10^{-4} \text{ mSv m}^2 / \text{h MBq} \times 555 \text{ MBq}}{(1 \text{ m})^2} \times 1 \text{ h/pat} \times 2080 \text{ pat/year} \times 0.64 \times 0.832 \times 1 = 85 \text{ mSv/year}$$

Consequently, the staff should develop procedures to minimize the time spent near the radioactive patient. Information collection, explanations, and blood collection or other tests should be performed as much as possible before radioactivity has been administered. Remote monitoring of the patients using video cameras can also be used to reduce the time technologists and nurses spend in close proximity to the patients. In other words, the practice should be optimized.

By applying formula (1), at the time the patient is being positioned for imaging, his initial activity is found to be

$$\begin{aligned} A(t_{up}) &= A_0 \cdot e^{-\lambda \cdot t_{up}} - \alpha \cdot A_0 = 555 \text{ MBq} \times \exp\left(-\frac{\ln 2}{109.8 \text{ min}} \times 60 \text{ min}\right) - 0.15 \times 555 \text{ MBq} \\ &= 297 \text{ MBq} \end{aligned}$$

The time for carrying the patient up to the acquisition suite and accommodate him in the correct position could be, conservatively, about 3 minutes, so that the dose to the technician from this operation would be

$$t_{op} = 3 \text{ min} \Rightarrow R_t(3 \text{ min}) \cong R_t(0) = 1$$

$$D = \frac{1.3910^{-4} \text{ mSv m}^2 / \text{h MBq} \times 297 \text{ MBq}}{(1 \text{ m})^2} \times \frac{3 \text{ h}}{60 \text{ pat}} \times 2080 \text{ pat/year} \times 0.64 \times 1 \times 1 = 2.75 \text{ mSv/year}$$

From the calculations carried out it is apparent that it is feasible for a single technician to exceed the dose constraint (5 mSv in the present case). The only reasonable way to lower this dose is to have enough staff so that the contact time between radioactive patients and any one staff member can be diluted.

The gantry and detectors of the PET tomograph can provide a substantial reduction of the dose rate at some of the walls. This depends on the actual geometry and placement of the tomograph in the room as well as the type of scanning procedures. If information on the tomograph shielding characteristics is available from the vendor, it can be incorporated into the calculation for the walls that are shielded by the scanner and for the floors and ceilings. Normally the most conservative approach is taken, i.e. no shielding from the tomograph is assumed. In this way, the calculation of shielding for the tomograph room is similar to the uptake area calculation. During the patient image acquisition, at least one technologist is located at the PET system console where both the patient and the progress of the imaging study can be monitored. Ideally, the console area should be located more than 2 m away from the scanner to reduce the operator dose below ALARA levels. Using formula (8) with these assumptions the dose to console operator from patient at PET scanner is found to be

$$D = \frac{1.3910^{-4} \text{ mSv m}^2 / \text{h MBq} \times 297 \text{ MBq}}{(2 \text{ m})^2} \times 1 \text{ h/pat} \times 2080 \text{ pat/year} \times 0.64 \times 0.832 \times 1 = 11.4 \text{ mSv/year}$$

In order to make a simple didactical calculation it can be assumed that patient at PET scanner is the only radioactive source. In this case, the transmission factor needed to comply with the 5 mSv/year dose constrain would be

$$k = \frac{5 \text{ mSv/year}}{11.42 \text{ mSv/year}} = 0.4378$$

which, according to table 4, corresponds to about 7 mm of lead ($k=0.3903$). Of course any combination of materials may be used to obtain the same value. For example if a layer of 2 mm of lead were used ($k=0.7873$), then the needed additional layer of concrete would be

$$k = 0.4378 = 0.7873 \times k_{conc} \Rightarrow k_{conc} = 0.5561$$

which corresponds to about 8 cm of concrete ($k=0.5180$). In other case, if masonry of mean density $\delta = 1.6 \text{ g/cm}^3$ is used, the equivalent thickness would be (see formula (3))

$$X_{brick} = \frac{2.35 \text{ g/cm}^3}{1.60 \text{ g/cm}^3} \times 8 \text{ cm} \cong 12 \text{ cm}$$

New facilities can efficiently use concrete to achieve required shielding factors, while in existing facilities, lead is often the best resort. Portable lead shields can be used effectively to shield patients in uptake rooms where they are required to remain stationary. Planning for new PET facilities should carefully consider the constraints associated with the regulatory limits. Uncontrolled areas with high occupancy should be located as far as possible from the uptake and imaging rooms. Also, the placement of the door must be carefully considered to avoid the expense with installing one with

substantial lead shielding. If uncontrolled areas are located above and below the PET uptake and tomograph rooms, the spacing between floors may need to be greater than normal. If that is not feasible, the floors need to be able support the weight associated with additional shielding. Floors often (but not always) have 10 cm of concrete, which will provide a dose reduction factor of 2.5. It is a good idea to have the hot bathroom, reserved for PET patients, within the immediate imaging area so that they do not alter the background counts of other detection devices as they pass through the clinic. The patient may be released immediately following the procedure or may go to a waiting area while the PET study is reviewed. If the patients are kept in the clinic for any length of time after the study is completed, that area must also be included in the radiation safety planning.

It should be noted that the lead shielding required in the walls for the CT system alone (i.e. ignoring the PET component) will have only a modest shielding effect for the 511 keV annihilation radiation. For example, 1.6 mm of lead will provide a transmission factor of only 0.81 for the annihilation radiation. Because the half value layer (HVL) for the CT x rays is so much smaller than that for 511 keV photons, a room that is shielded to meet the general public levels for PET usually is unlikely to need additional shielding for the CT component.

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Current Situation of the Facilities, Equipments and Human Resources in Nuclear Medicine in Argentina

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CURRENT SITUATION OF THE FACILITIES, EQUIPMENTS AND HUMAN RESOURCES IN NUCLEAR MEDICINE IN ARGENTINA

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Abstract. The current situation of nuclear medicine in Argentina, taking into account the facilities, their equipment and human resources available is presented in this paper. A review and analysis of the equipment, including technical characteristics and a survey of the professionals and technicians of the area, was carried out. In Argentina, there are 266 centers of nuclear medicine distributed all over the country. The operating licenses are granted by the Nuclear Regulatory Authority (Autoridad Regulatoria Nuclear – ARN). Forty four percent of the installed equipment are SPECT of 1 or 2 heads and 39,4 % are gamma camera. Besides, there are eleven PET operating in Argentina. There are 416 nuclear medicine physicians with individual permit for diagnostic purposes and 50% of them has also individual permit for treatment purposes. With the purpose of analyzing the regional distribution of the available resources in nuclear medicine, the country was divided into 7 geographical regions: City of Buenos Aires, Province of Buenos Aires, Pampa, Cuyo, Northeast, Northwest and Patagonia. From the analysis of the gathered information it is possible to conclude that the nuclear medicine equipment as well as the personnel present an irregular distribution, with a major concentration in the City of Buenos Aires and Province of Buenos Aires. The Northeast region presents the lowest number of Nuclear Medicine centers and the Patagonia region has the lowest number of medicine nuclear physicians with individual permits. The number of SPECT and gamma cameras is 7,3 per million of inhabitants. The information about the available resources in nuclear medicine presented in this paper and its comparison with the international information available provide elements for a better planning of the future activities in the area not only for the operators but also from the regulatory point of view.

Keywords: nuclear medicine, equipment, personnel, Argentina

Since the end of the sixties, Nuclear Medicine in Argentina has progressed in an increasing and accelerated rhythm. Several centers have been installed, a huge number of specialists have been trained, the number of courses recognized by the Argentinean Nuclear Regulatory Authority (Autoridad Regulatoria Nuclear - ARN) that provide theoretical as well as the practice required not only in medicine but in the area of teaching and research has also increased and several congresses, conferences, symposiums, and meetings have been developed on this matter.

The development of new techniques and the equipments have followed the same course with an increase in number and diversity of studies and treatments, while research is open to new perspectives of application.

In Argentina, there are 266 nuclear medicine centres at present that possess operation license granted by RNA, distributed around the whole country. This license varies according to the equipment the facility possesses and the purpose of use of the radioactive material.

With the aim of getting to know the present situation in the area, an evaluation of the characteristics of the facilities, their equipment and personnel was carried out.

The data were obtained from ARN records. ARN grants operation licenses to use unsealed sources in nuclear medicine after the evaluation of the documentation handed in by the solicitor and the audit in which the inspectors determine that the facility complies with the requirements of Standard AR 8.2.4 “Use of unsealed radioactive sources in nuclear medicine”. (1)

The titular of a license must appoint a person responsible for the radiological safety who must have an individual permit for the same purpose of use of the installation. The responsible must elaborate and implement the quality system of the nuclear medicine service among other things and is the interlocutor between the facility and the ARN. The data existing in ARN databases are checked and/or

modified during the regulatory inspections that take place in the facilities with a frequency of at least every two years or when ARN considers proper.

The information about the current situation of nuclear medicine installations in Argentina, presented in this paper was, is taken from an análisis of the records available in ARN.

In Table 1 the distribution of nuclear medicine services in Argentina is shown. For this purpose the country has been divided into 7 geographical regions. Besides, the number of nuclear medicine services per million of inhabitants is shown, reaching an average of 7,3 per million of inhabitants in the country. In Argentina 10,3 % of nuclear medicine services corresponds to public institutions and the remaining 89,76 % to private institutions.

In Table 2, type of equipment existing in each service is detailed and compared to the equipment existing during the period between 1991-1996 (2)

Table 1: Nuclear Medicine Services in Argentina according to geographical distribution.

Region	N° of inhabitants (3)	Surface for Km²	N° of Nuclear Medicine Services	Serv./million of inhabitants
City of Buenos Aires	2.776.138	200	58	20,89
Prov. of Buenos Aires	13.827.203	307.571	103	7,45
Pampa	7.524.943	520.549	50	6,64
Cuyo	2.567.607	315.226	15	5,84
Northeast	3.367.518	289.699	10	2,97
Northwest	4.458.470	559.864	19	4,26
Patagonia	1.738.251	1.768.165	11	6,33
Total	36.260.130	3.761.274	266	7,33

Figure 1: The regions are constituted by the provinces detailed as follows:

Pampa: Córdoba, Santa Fe, Entre Ríos, La Pampa.

Cuyo: San Juan, Mendoza, San Luis.

Northeast: Chaco, Formosa, Misiones, Corrientes.

Northwest: Salta, Jujuy, Tucumán, Catamarca, La Rioja, Santiago del Estero.

Patagonia: Río Negro, Chubut, Santa Cruz, Tierra del Fuego, Neuquen, Antártida Argentina e islas del Atlántico Sur.



Table 2: Equipment of nuclear medicine installed in Argentina, comparison between periods 2002-2007 and 1991-1996

Nuclear Medicine Equipments	Period 2002-2007	Period 1991-1996
Rectilinear Scanners	80	122
Planar Gamma Camera + SPECT	434	311
SPECT Double head	27	-
PET Scanners	11	1

With the purpose of analyzing the territorial distribution of equipment availability, the number of rectilinear scanners, planar gamma cameras, one or two head SPECT and PET are presented according to their distribution in the different regions.

From the data analysis, an irregular distribution of nuclear medicine centers is observed as well as the concentration of the equipment in the City of Buenos Aires and the Province of Buenos Aires.

Table 3: Number of nuclear medicine equipments per geographical region.

Region	Rectilinear Scanners	Planar Gamma Camera	SPECT	SPECT Double Head	PET
City of Bs.As	15	28	44	11(*)	7
Buenos Aires	10	43	60	7	3
Pampa	9	11	40	5	-
Cuyo	7	5	9	2	1
Northeast	4	4	10	-	-
Northwest	4	10	10	-	-
Patagonia	-	5	7	1	-
Total	49	106	180	26	11

(*) One of the 2 head-SPECT operates as a PET.

According to the characteristics of the duties that take place at nuclear medicine services, their equipment and personnel, the facilities may ask ARN an operation license only with the purpose of use as radioactive tracers for dynamics studies (diagnosis) or for treatments such as hyperthyroidism, thyroid carcinoma with I-131, heart disfunctions, pathologies palliation of painful bone metastases using Sm-153, Sr-89 or P-32.

The number of nuclear medicine services operating in Argentina taking into account the purpose of use of the radioactive material is shown in Table 4.

Table 4: Distribution of nuclear medicine services in Argentina per purpose and region.

Region	Total N° of services	Quantity per purpose	
		Only diagnosis	Diagnosis and treatment
City of Buenos Aires	58	18	40
Province of Buenos Aires	103	43	60
Pampa	50	23	27
Cuyo	15	6	9
Northeast	10	6	4
Northwest	19	9	10
Patagonia	11	5	6

Regarding the quantity of personnel in a nuclear medicine installation, Standard AR 8.2.4 “Use of unsealed radioactive sources in nuclear medicine” establishes that it must be set taking into account the studies or practices that take place in a given service, the equipment available and the work load.

The minimum staff must be:

- a) Enough doctors with individual permit to make the corresponding studies or treatments to cover the whole schedule when radioactive material is administered to patients.
- b) Professional personnel or technicians with individual permits with proper qualification to manipulate radioactive material, in an adequate number corresponding to the work load of the nuclear medicine service.

In case that high complexity nuclear medicine equipment is used, such as PET scanners or double head SPECT, to perform coincidence measures, the staff must also count on a part time specialist in health physics. There are 349 physician in Argentina with individual permit to use radioactive material in nuclear medicine. The distribution of physician with individual permits that could be appointed responsible for radiation safety in any nuclear medicine centre, according to their geographical area of residence is shown in Table 5.

Table 5: distribution of doctors in nuclear medicine with individual permits per region and purpose.

Region	Only diagnosis	Diagnosis and treatment (*)	Diagnosis and treatment (**)
City of Bs.As.	116	66	35
Buenos Aires	100	112	11
Pampa	64	26	6
Cuyo	21	12	1
Northeast	11	3	-
Northwest	24	13	1
Patagonia	13	6	1

(*)**Diagnosis and treatment** with I-131 for hyperthyroidism and thyroid carcinoma.

(**)**Diagnosis and treatment** with I-131 for hyperthyroidism and thyroid carcinoma and palliation of painful bone metastases with Sm-153, Sr-89 and P-32

In relation to nuclear medicine technicians, there are 101 technicians Experts that possess individual permits. There are 34 in the City of Buenos Aires, 55 in the Province of Buenos Aires, 5 in Cuyo, 6 in Pampa Region, and 1 in Northeast Region.

During the last years it was noticed an increasing tendency in the number of individual permits applications on the part of nuclear medicine technician due to a greater offer of specific theory training courses and the demand to comply with the rules in force.

From the evaluation of the gathered data the conclusion is that human and material resources in the frame of nuclear medicine present regional irregular distribution, with an important concentration in the City of Buenos Aires and the Province of Buenos Aires. Northeast region is the one with the lowest number of installed nuclear medicine services and Patagonia is the region with the lowest number of professionals living in the area. This situation sometimes forces other radiological safety responsables to come over from other regions and, consequently, that the services can not operate every day. The average number of measuring equipment (Gamma Cameras and SPECT) per million of inhabitants reaches 7,3 within a range of values between 3,0 equipments per million operating in the Northeast Region to 20,9 in the City of Buenos Aires, while the European Community recommends 10 Gamma Cameras per million of inhabitants.

The information recollected and presented in this paper has proved to be very useful, because it can be used to evaluate the available resources and compare them to international data. Besides, it contributes to a rationalization of human and material resources and it could eventually be the basis for the planning of future activities in the area of nuclear medicine from the operators' point of view of as well as from the competent Nuclear Regulatory Authority.

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ARN Results in Interlaboratory Comparison Exercises

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ARN RESULTS IN INTERLABORATORY COMPARISON EXERCISES

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Abstract. For years, the Nuclear Regulatory Authority (ARN) has been involved in several laboratory intercomparison programmes. The objective of participating in these exercises is to assure the quality of the determinations that the radiochemical laboratories of ARN carries out as part of its regulatory activity. Most of these determinations are related to its environmental monitoring program in the vicinity of nuclear and radioactive facilities existing in the country, in operation or not. Other determinations are related with effluent samples and monitoring activities performed inside the facilities.

On the other hand, these intercomparisons are part of the requirements for the laboratories under ISO 17025. ARN laboratories are in process to obtain or maintain ISO 17025 accreditation as a priority objective.

During the development of the intercomparisons, different samples have been tested in several matrices containing alpha, beta and gamma emitters. These exercises were organized by different laboratories as the IAEA, the EML and NIST from United States, the NPL and the NRPB from England, the BFS from Germany, and so on.

The results were very satisfactory not only in direct measurements (gamma spectrometry) but also in those that require a previous intensive laboratory processing (alpha spectrometry and liquid scintillation), resulting in many cases better than the general average. This paper provides a summary of the results obtained in these exercises and the results are compared with the overall average of the participating laboratories.

KEYWORDS: *Interlaboratory, inter comparison, ARN laboratories, performance*

1. Results obtained by ARN in the intercomparison exercises during 2007

During 2007 the ARN laboratories performed several intercomparison exercises organized by laboratories from different countries. Table 1 shows a summary of these exercises with their corresponding results:

Table 1: Summary of 2007 exercises with ARN's results

Organization	Exercise	Matrix	Type analysis	Number of results	Assessment ^(*)
National Institute of Standards and Technology (NIST), USA	"NIST Radiochemistry Intercomparison Program, NRIP'07" ^[1]	Water	H-3, uranium and gamma emitters	5	A: 80,0 %
					N: 20,0 %
Health Protection Agency (HPA), UK	"2007 Intercomparison of passive radon detectors"	Air collected on Activated charcoal	Rn-222	5	Not evaluated

National Health Laboratory (NPL), UK	"NPL ENVIRONMENTAL RADIOACTIVITY PROFICIENCY TEST EXERCISE 2007" ^[2]	Water	Alpha, beta and gamma emitters	38	A: 84.2 %
					W: 10.5 %
					N: 5.3 %
International Atomic Energy Agency (IAEA)	"IAEA CU-2007-09 Proficiency Test On the Determination of Po-210 In Spiked Water" ^[3]	Water	Po-210	4	A: 100 %
International Atomic Energy Agency (IAEA)	"The IAEA-CU-2007-03 World-wide open proficiency test" ^[4]	Water, soil and vegetal	Alpha, beta and gamma emitters	24	A: 87.5 %
					W: 4.2 %
					N: 8.3 %
Comisión Nacional de Energía Atómica (CNEA), Argentina	INTERLAB RU-1 ^[5]	Water	Uranium	6	A: 100 %
Total of results:				82	

(*) **A** = "Accepted", "traceable", "in agreement", etc.

W = "Accepted with warning", "questionable", etc.

N = "Not accepted", "discrepant", etc.

according to the exercise organizer.

In summary, the intercomparison exercises resulted in the following general averages (Fig. 1):

Figure 1: 2007 - ARN's general averages.

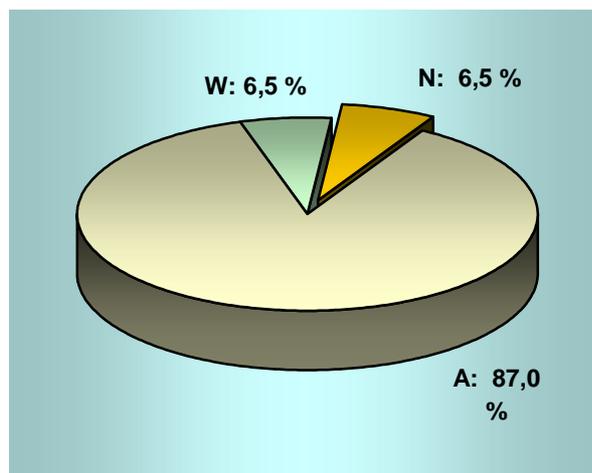
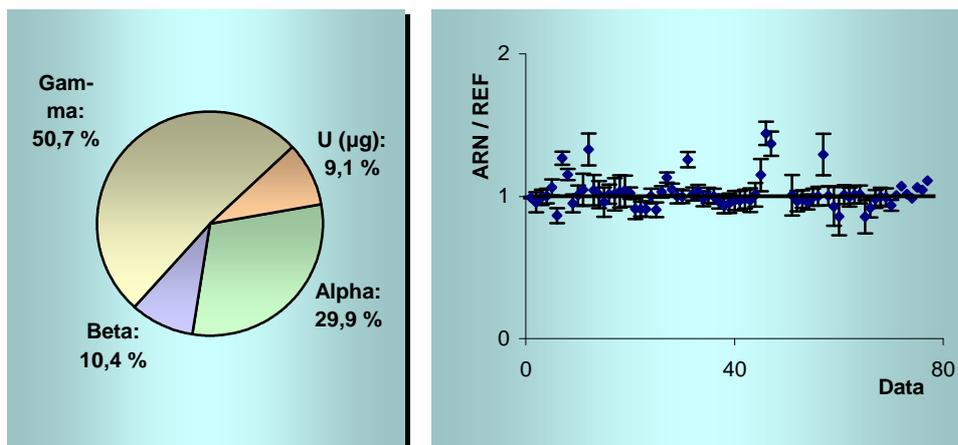


Figure 2 shows the percentage distribution of the reported results, presented by type of emission: gamma emitters (direct measurements), alpha and beta emitters (indirect measurements). Figure 3 presents a distribution of all the results in relation with the reference values. It is also shown the difference between each result and the corresponding reference value.

Figure 2: Percentage distribution of the reported results, presented by type of emission: gamma emitters, alpha and beta emitters (Left) and **Figure 3:** Distribution of all the results in relation with the reference values (Right).



In the figure 2, the sum of the percentages is not 100% due to the rounding.

Table 2 shows the radionuclides analyzed by type of emission, as well as the evaluated results achieved in each category, and the total reported results for each group of emitters.

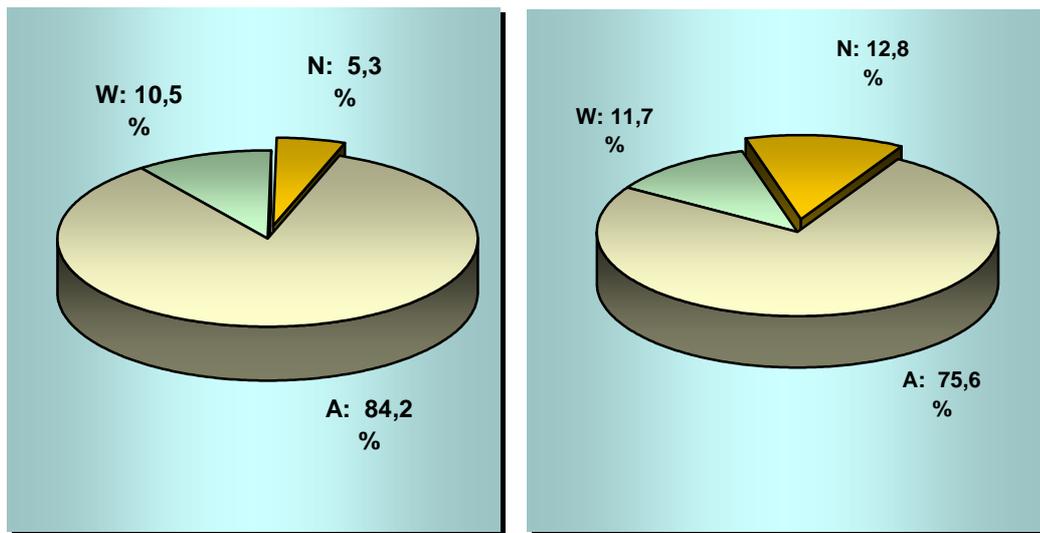
Table 2: Radionuclides analyzed by the ARN, by type of emission.

Type of emission	Radionuclides	A %	W %	N %	Total of results
Alpha	Ra-226 U-238 Pu-238 Pu-239 Am-241 Cm-244 Po-210	87.0	4.3	8.7	23 (+5 not evaluated)
Beta	Ni-63 Sr-90 H-3	50.0	37.5	12.5	8
Gamma	Co-60 Zr-95 Nb-95 Sb-125 Ba-133 Cs-134 Cs-137 Ce-144 Eu-152 Eu-155 K-40 Cd-109 Mn-54 Pb-210 Zn-65 Co-57	94.9	2.6	2.6	39
U (µg)	U	85.7	0	14.3	7

2. Results obtained by ARN in the intercomparison exercise organized by the NPL (2007)

The 2007 NPL report is the most complete and it allows comparing ARN results with all participating laboratories. In Figure 4 are shown the results obtained by ARN and in Figure 5 the average results obtained for all the laboratories in this exercise, being “non reported” gamma isotopes excluded.

Figure 4: NPL 2007- Results obtained by ARN (Left) and **Figure 5:** NPL 2007- Results obtained by all laboratories (Right).

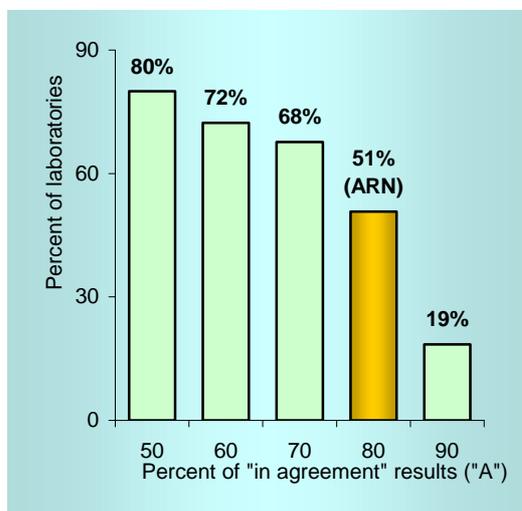


In the figure 5, the sum of the percentages is not 100% due to the rounding.

The results described as “in agreement” ranged from 25% to 100%. As it can be seen, the average of all laboratories for grade “A” was 75.6%.

Figure 6 shows the number of laboratories that exceed a certain percentage of accepted results. It is important to point out that ARN is one of the 51% of participating laboratories that achieved more than 80% of accepted results. Only 19% of the laboratories had more than 90% of accepted results.

Figure 6: NPL 2007- Laboratories with a percent of accepted dates better than...



When the participating laboratories are sorted by the percentage of results “in agreement”, it is important to remark that the laboratories of ARN are located in position number 27 when the total number of participating laboratories was 65.

3. Results obtained by ARN in several intercomparison exercises prior to 2007

Table 3 shows an overview of the intercomparison activities carried out by ARN’s laboratories from 1994 until 2006:

Table 3: Overview of the intercomparison activities of the ARN's laboratories from 1994 until 2006.

Year	Organism	Exercise	Matrix	Type analysis	Number of results	Assessment(*)
1994	Laboratoire D'Analyses de Biologie Medicale de Commissariat a L'Energie Atomique (CEA), France	"Radiological Intercomparison Exercise - Quality Control Tritium in Urine and water" ^[6]	Water and urine	H-3	5	A: 100 %
1995 to 2004	Environmental Measurements Laboratory (EML), USA.	"Quality Assessment Program (QAP)" ^[7-9]	Water, soil, vegetables and filters	Alpha, beta and gamma	718	A: 82.9 %
						W: 13.4 %
						N: 3.8 %
1994, 1995 and 1996	Environmental Measurements Laboratory (EML), USA	-	Scintillation cells	Rn-222	12	Not evaluated
1998	American Association of Radon Scientists and Technologists (AARST), USA	-	Scintillation cells, Activated charcoal and Electrets	Rn-222	14	Not evaluated
1999	Bundesamt für Strahlenschutz-Federal Office for Radiation Protection (BFS), Germany	BFS-RV-1999-Am-241 ^[10]	Urine	Am-241	1	A: 100 %
1999	Environmental Measurements Laboratory (EML), USA	"Eml gamma spectrometry data evaluation program"	-	Software of gamma spectrum analysis	-	Not evaluated
2000	Bundesamt für Strahlenschutz-Federal Office for Radiation Protection (BFS), Germany	BFS-RV-2000-Pu-240 ^[11]	Urine	Pu-240, U-238 and U-234	3	A: 100 %
2001	International Atomic Energy Agency (IAEA)	"Alpha emitters in urine samples"	Urine	Alpha emitters	11	Not evaluated

2003	Bundesamt für Strahlenschutz-Federal Office for Radiation Protection (BFS), Germany	BFS-RV-2003-Th/U	Urine	Uranium and thorium	10	A: 80 %
						N: 20 %
2003	National Radiological Protection Board (NRPB), UK	-	Activated charcoal and Electrets	Rn-222	15	Not evaluated
2004	Environmental Measurements Laboratory (EML), USA	“Eml gamma spectrometry data evaluation program”	-	Software of gamma spectrum analysis	-	Not evaluated
2005	Health Protection Agency (HPA), UK	“2005 Intercomparison of passive radon detectors”	Activated charcoal	Rn-222	10	A: 100 %
2005	Bundesamt für Strahlenschutz-Federal Office for Radiation Protection (BFS), Germany	BFS-RV-2005-H-3 and BFS-RV-2005-Sr-90	Urine	H-3 and Sr-90	4	A: 75 %
						N: 25 %
2006	International Atomic Energy Agency (IAEA)	“The IAEA-CU-03 2006 worldwide proficiency test (PT)”	Water, grass and soil	Gamma emitters	18	A: 83.3 %
						W: 5.6 %
						N: 11.1 %
2006	National Institute of Standards and Technology (NIST), USA	“NIST Radiochemistry Intercomparison Program, NRIP'07”	Water	H-3, uranium and gamma emitters	6	A: 66.7%
						N: 33.3 %
Total of results:					827	A: 89.0 %
						W: 2.1 %
						N: 8.9 %

(*) **A** = “Accepted”, “traceable”, “in agreement”, etc.
W = “Accepted with warning”, “questionable”, etc.
N = “Not accepted”, “discrepant”, etc.
according to the exercise organizer.

Combining the values calculated for 2007 and those that come out from the above table, the following values are obtained for the period 1994-2007 produced from 909 results (Table 4):

Table 4: Combining averages for ARN from 1994 until 2007.

A (%)	W (%)	N (%)
88.0	4.3	7.7
Total of results:		909

4. Conclusions

In general, the performance of the ARN laboratories was very satisfactory not only in direct measurements (gamma spectrometry) but also in those that require a previous intensive laboratory processing (alpha spectrometry and liquid scintillation), resulting in many cases better than the general average.

During 2007, the laboratories of ARN obtained 87.0% of accepted results, with 82 reported data (and 77 evaluated data) in 6 intercomparison exercises.

In the participation in the exercise organised by the NPL, the ARN reached 84.2% of accepted results, being one of the 51% of participating laboratories that achieved more than 80% of accepted result. The average result achieved for grade “A” was better than the average of all laboratories, which was 75.6%.

Since 1994 the ARN has participated in 40 intercomparison exercises and has produced 909 results with 88.0% of them qualified as grade “A”.

These results confirm the quality of the determinations carry out by the ARN laboratories as part of its regulatory activities.

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Tritium Determination in Water

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TRITIUM DETERMINATION IN WATER

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Abstract. An analytical procedure for the determination of tritium in water is described in this paper. The determination is carried out in presence of other radionuclides, such as Fe-55, Ni-63, Mn-54, Zn-65, Co-60, Cd-109, Sr-90, Cs-134 and Cs-137. The method consists in a simple distillation stage prior to measurement by liquid scintillation counting. The samples containing beta and gamma emitters are conditioned with a $(\text{NO}_3)_2\text{Pb}$ solution and Na(OH) up to pH = 7 - 8. This produces lead hydroxide precipitation that allows fixing volatile elements, which could be transported together with tritium, and may increase the extinction degree of the sample or interfere with the counting process. Special attention must be paid if presence of Fe-55 ($E_{\text{max}} \sim 5.95$ keV) is suspected as it might not be distinguished from tritium ($E_{\text{max}} \sim 18$ keV), leading to an overestimation of tritium activity.

Different tests were carried to obtain the optimum method conditions, to achieve the purification of the tritium and a pH near to 7 in the distilled. The detection limit (2σ) was 8.0 Bq/l and the distillation performance was 98.3 %.

This technique was applied to water samples containing Fe-55 and other gamma radionuclides in 1M hydrochloric acid media in successive Environmental Measurements Laboratory (EML), U.S. Department of Energy (DOE) intercomparison programs. The results obtained were very satisfactory and are presented in this paper.

KEYWORDS: *distillation, tritium, method.*

1. Introduction

Tritium (H-3) is mainly produced in PHWR reactors by neutronic activation of deuterium of heavy water moderator and coolant system. During normal operation some of the H-3 is released into the environment through liquid and gaseous discharges. Nuclear Regulatory Authority (ARN) performs analysis of H-3 on environmental samples from superficial and ground waters and also samples from liquid and gaseous discharges of nuclear power plants.

When large amounts of beta and gamma emitters are present in the samples, they could interfere in the H-3 measurement and it becomes necessary to distillate the sample to remove them. Under certain conditions, some elements could be transported to the distillate, especially when the distillation process reaches its final stage (dry), resulting in an inefficient separation. For example, when Fe is in hydrochloric acid media, the alkalization of the sample does not totally prevent Fe from passing to the distillate. However, the presence of Fe-55 in the distillate interfere in the H-3 measurement, due to its low energy ($E_{\text{max}} \sim 5.95$ keV). Therefore it would be overlapped in the H-3 spectrum, increasing erroneously the counting.

The objective of this paper is to optimize the distillation process in order to achieve an adequate separation of H-3 from other volatile elements, adjusting work conditions in order to obtain a distillate with an adequate pH (within 7).

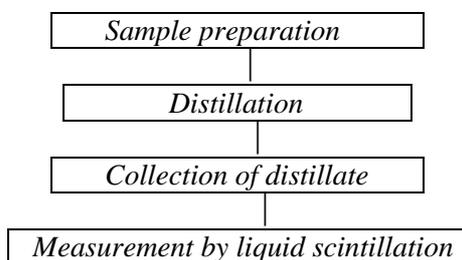
Several samples arising from the intercomparison exercises programs organized by EML contained some beta emitters. These samples were submitted to the distillation process and the absence of beta emitters was then confirmed in the distillate.

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2. Experimental stage

Tritium determination was carried out by following the scheme shown in Figure 1. The samples were finally measured in a Perkin Elmer Tri Carb 2770 TR / SL liquid scintillation equipment.

Figure 1: Scheme of the complete procedure



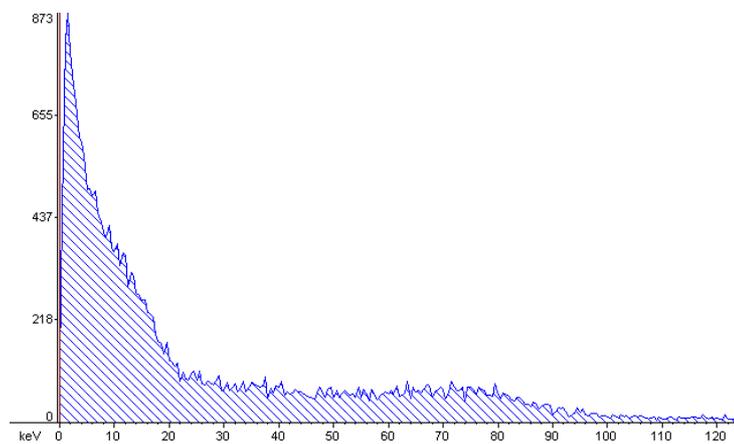
A batch of samples was prepared in 1M HCl media containing known and similar activities of H-3 and Fe-55. After that, the addition of 11,6 g of 5M Na(OH) and 20 g of Pb(NO₃)₂ was carried to reduce the alkali excess, forming a Pb(OH)₂ white precipitate. Finally, the distillation was performed up to dryness. The endpoint was indicated by the formation of PBO, a yellow precipitate [1-2].

Once the distillate was obtained, the final liquid was weighted and a 10 g aliquot was taken into a polyethylene vial containing 10 ml of Insta Gel XF as scintillation cocktail. The vials were left in darkness during 30 minutes and then measured during 150 minutes according to a predetermined protocol.

3. Measurements by liquid scintillation counting

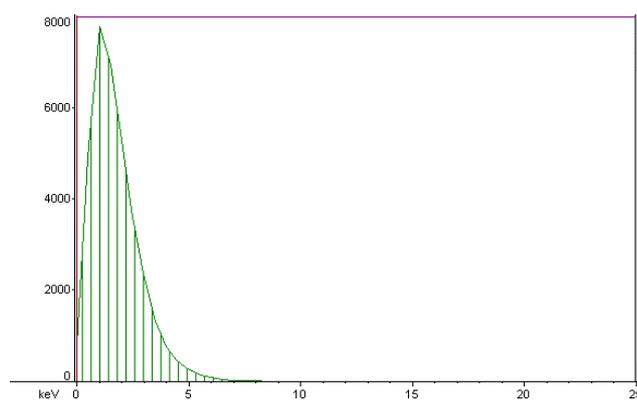
The sample spectrum (window from 0 to 120 keV) containing H-3 and several beta and gamma emitters (such as: Fe-55, Ni-63, Mn-54, Zn-65, Co-60, Cd-109, Sr-90, Cs-134 y Cs-137) is shown in Figure 2.

Figure 2: Spectrum obtained in a direct measurement of an intercomparison water sample



After the distillation, the sample spectrum can be observed in Figure 3.

Figure 3: Spectrum obtained by liquid scintillation, with previous distillation, of the same intercomparison water sample.



The results of H-3 recovery assays and the Fe-55 retention tests are shown in Table 1:

Table 1: Retention of H-3 and the retained amount of F-55 in the distillations

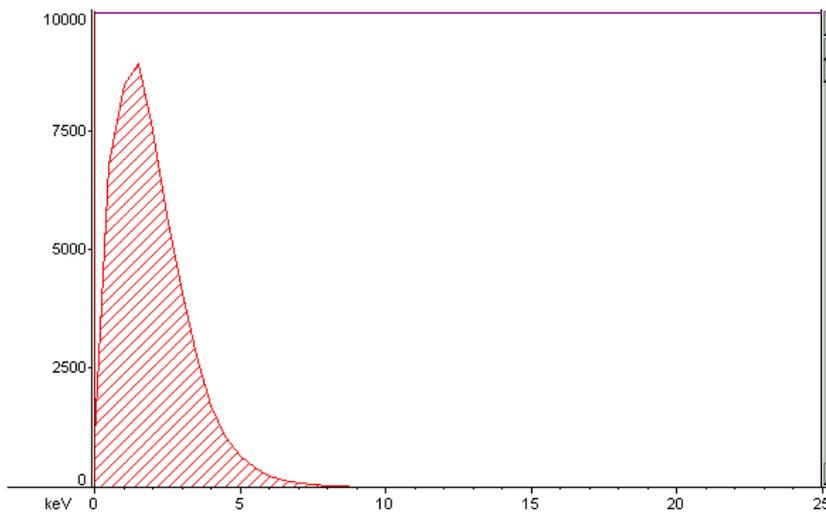
Number	H-3 Recovery (%)	Fe-55 Retention (%)
1	96.27	104.43
2	100.87	98.33
3	97.66	98.68
4	100.44	92.79
5	96.56	96.72
6	101.75	103.21
7	95.62	100.59
8	95.75	93.97
9	96.50	100.04
10	101.80	100.96

The average obtained was 98.3 % for tritium retention, with a $\sigma = 2.6$ %, whereas for Fe-55 retention, the average result was 99 % with a $\sigma = 3.7$ %.

In order to determine the distillation efficiency, several parameters were controlled, such as distillation time, boiling time, appearance of first drop of distillation and final volume collected.

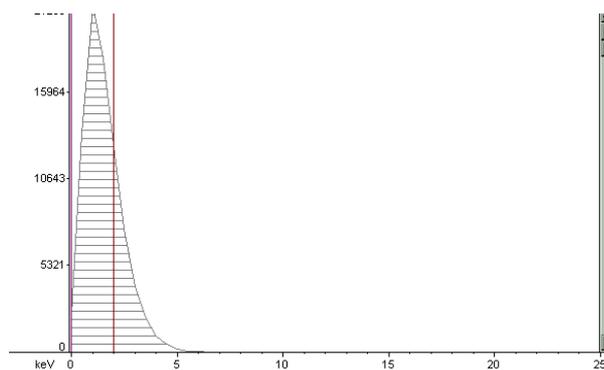
Figure 4 shows one of the spectra obtained in the recovery tests.

Figure 4: Distillation efficiency. Spectrum obtained according to the specified work conditions.



In Figure 5, it is shown a spectrum of Fe-55 (distillate waste), after the distillation of the sample, according to the mentioned conditions.

Figure 5: Fe-55 retained in the distillation flask. Spectrum obtained according to the work conditions.



The initial distilled quantity was 50 g. The measurements were performed in a Perkin Elmer Tri Carb 2770 TR / SL Packard Liquid Scintillation Analyzer Equipment, with a measure efficiency of 20 % in the 0 - 5 Kev region. Finally, a curve of extinction with 10 points was performed in order to determine the activity.

4. ARN results in the Quality Assurance Program of EML (QAP-DOE)

The technique described above was applied to samples from the Quality Assurance Program of EML (DOE), in which the ARN participated from 1995 to 2004, with a frequency of two sets of samples per year. In this QAP program, more than 130 laboratories had participated, and the information was classified in three categories: Class A or Accepted; Class W or Accepted with reservations; and Class C as non-accepted. In table 2 are shown the results obtained by ARN in 15 intercomparison exercises. The EML references values are also shown [3 - 14].

Table 2: Results obtained by ARN in intercomparison exercises performed by EML.

QAP (*)	ARN (Bq / l)	EML (Bq / l)	Evaluation
44	226.0 ± 5.0	251.0 ± 11.4	A
45	500.0 ± 30.0	587.0 ± 58.0	A
47	122.2 ± 5.0	115.0 ± 6.0	A
48	213.8 ± 10.7	218.3 ± 6.5	A
49	76.9 ± 1.5	76.2 ± 2.9	A
50	115.8 ± 1.2	121.1 ± 6.8	A
51	77.1 ± 1.5	80.7 ± 3.7	A
52	85.7 ± 2.2	79.4 ± 2.5	A
52	85.6 ± 2.4	79.4 ± 2.5	A
52	85.3 ± 2.0	79.4 ± 2.5	A
53	84.8 ± 2.0	91.3 ± 0.3	A
53	86.0 ± 2.0	91.3 ± 0.3	A
54	66.6 ± 1.5	79.3 ± 2.0	A
56	283.6 ± 8.6	283.7 ± 3.8	A
57	250.2 ± 5.3	227.3 ± 5.7	A
58	396.6 ± 7.9	390.0 ± 3.4	A
59	458.8 ± 9.2	446.3 ± 2.2	A
60	253.6 ± 10.2	186.6 ± 3.3	W

(*) Quality Assessment Program. Number given by the EML to each intercomparison exercise.

In the last exercise it was obtained a questionable result (W), due to the aging of the scintillation solution. For that reason, and due to the fact that the scintillator was discontinued in its fabrication, it was replaced by another one, Ultima Gold LLT. The results were corrected and validated through the participation in two intercomparisons with the National Institute of Standard and Technology (NIST).

5. Conclusions

From the information reported in table 1, it can be concluded that the method presented in this paper fulfill laboratory requirements as obtained results are optimal. Moreover, this is a simple, quick and economic technique.

In relation with the data presented in Table 2, it can be seen that results obtained by ARN in the past 10 years were very satisfactory. The values were very similar to EML reference values.

The participation in these intercomparison exercises allowed to assure the quality of the determinations performed by the tritium laboratory of ARN.

Acknowledgements

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^{222}Rn Measurements in Dwellings of Argentina

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^{222}Rn measurements in dwellings of Argentina

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Abstract. Radon gas (^{222}Rn) is responsible of about fifty per cent of the world population dose due to natural sources, being the most important pathway the inhalation of radon progeny, specially indoors. Radon concentration has been measured in dwellings at different locations in Argentina. The places selected to be evaluated are representative of the different geologic zones of the country. Near 3000 dwellings have been analyzed since 1983 up today. The measuring methods used in this case were track etched detectors, electrets and detectors based on activated charcoal adsorption. Two different methods with track etched detectors were used: a simple one, which determines only the average radon concentration, and a second one that measures both radon concentration and the equilibrium factor (F) between radon and its daughters. The last one is a method that uses two Makrofol passive track detectors in the same device. The average radon concentration value obtained from all the dwellings evaluated was $44.6 \text{ Bq}\cdot\text{m}^{-3}$. The annual effective dose calculated from this average concentration, using a dosimetric factor of $25 \mu\text{Sv}\cdot\text{a}^{-1}(\text{Bq}\cdot\text{m}^{-3})^{-1}$, which assumes an equilibrium factor of 0.4, was 1.121 mSv. The average value obtained from the 222 dwellings evaluated by the second method was $49.3 \text{ Bq}\cdot\text{m}^{-3}$ and 0.37 the equilibrium factor, resulting the annual effective dose estimated 1,44 mSv. The measured equilibrium factor of 0.37 allows us to verify the assumed equilibrium factor of 0.4. Finally, radon levels in dwellings of Argentina are within the acceptable values for population, not being necessary to implement remedial actions, except in isolate cases that are still under study.

KEYWORDS: Radon, natural radioactivity, indoor radon, dwellings.

1. Introduction

In the last years, exposure of the population to natural radiation sources and especially to radon and its daughters has become increasingly important among members of the public and national authorities. Radon gas is responsible of about fifty per cent of the world population dose due to natural sources, being the most important pathway the inhalation of radon progeny, specially indoors [1].

Radon gas is a noble gas of natural origin, member of the ^{238}U chain. Radon is released from soils and minerals and migrates into the atmosphere. Its half-life (3.81 days) is long enough to allow wide distribution and accumulation in the environment. Indoors, radon gas is accumulated and may reach significant concentrations. This can occurs both in dwellings and in workplaces, resulting in a significant natural source of exposure to humans [2].

Radon decays in the so called radon daughters, the short-lived radon products. The overall effective half-life of the four daughters (^{218}Po , ^{214}Pb , ^{214}Bi and ^{214}Po) is about 30 minutes. Radon gas and its progeny are related through their equilibrium factor F [3]. The dose due to radon exposure is mainly caused by the inhalation of these decay products and their deposition. The progeny become attached to the ambient aerosol, is inhaled and the radiation dose is delivered to the bronchial tissue, being associated with bronchogenic carcinoma at high concentrations [4].

Since the 80s, the ARN carried out a radon gas measurement program in dwellings of different Argentine cities, having measured near 3000 dwellings up to the date. The main objective of the measuring program was to determine radon concentrations and the equilibrium factor, to calculate the corresponding effective doses and to determine regional variations and detect dwellings with high radon concentrations.

The result of radon gas measurements and equilibrium factor carried out in Argentina are presented in this work, together with the corresponding effective annual dose levels estimated from the measured values.

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2. Materials and methods

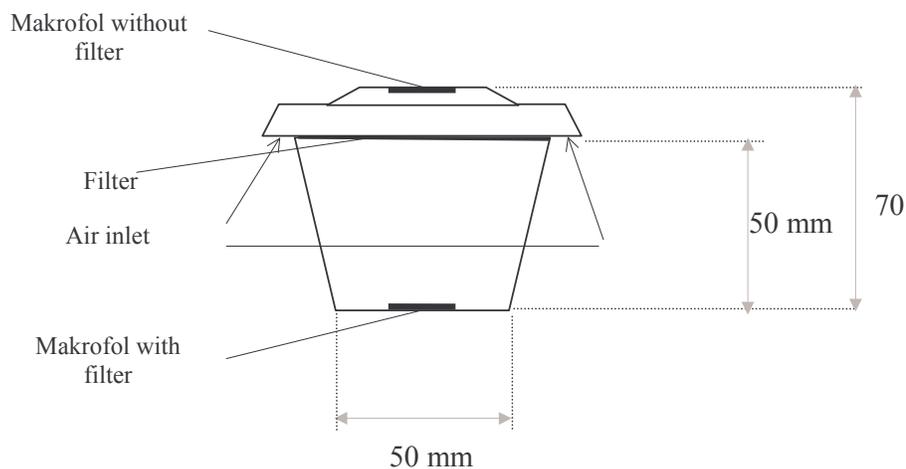
The places chosen for the monitoring program are representative of different geological zones present in Argentina: uraniferous, mountainous, valley and sedimentary area. The analysed dwellings have been built with different materials, but mainly brick and concrete, most used materials in our country. Different methods used to measure radon employ the detectors shown in Fig.1.

Figure 1: Nuclear track detectors, electret and activated charcoal detectors



As screening methods, electret detectors [5] and detectors based on radon adsorption on activated charcoal with subsequent measurement in a liquid scintillation counter were used [6]. As time integrated measurements, two solid state nuclear track detectors were employed. These detectors can be exposed between two and three months. [7-11]. One of the nuclear track detectors uses an acrylic material called CR-39 and determines only the average radon concentration during the exposure period of time. The other one is a polycarbonate material named Makrofol type E, and measures both radon concentration and the equilibrium factor (F) between radon and its daughters, since it uses two Makrofol sheets in the same device, as it is shown in figure 2.

Figure 2: Makrofol monitor used for determining radon gas concentration and equilibrium factor F



This method was developed by the ARN radon laboratory in order to determine the equilibrium factor value, which otherwise is assumed as 0.4 [3]. This value of 0.4 has been recommended by international organizations [4]. The measured equilibrium factor allows to perform a better estimation of the dose due to radon. Once finished the exposure period, the Makrofol sheets are revealed with an electroetching process while CR-39 is revealed by chemical etching and evaluated in a microscope.

The time integrated method was the most used because it gives a more representative value of the radon average concentration. In addition, these detectors are resistant and inexpensive, both conditions necessary for campaigns where a large number of dwellings have to be evaluated.

All devices were calibrated in a 1 m³ acrylic reference chamber where controlled conditions of humidity, temperature and radon concentrations can be maintained.

If only ²²²Rn gas concentration is known, the annual effective dose (E) is estimated as the product between gas concentration in air (C_{Rn}) and the dose conversion factor (f);

$$E = C_{Rn} f \quad (1)$$

where f is equal to 25 μSv.a⁻¹ / Bq.m⁻³ for 7000 hours per year and assuming F as equal to 0.4 [1]. When the equilibrium factor (F) is also known, the annual effective dose can be estimated by the following equation:

$$E = C_{Rn} (d_0 + d_1 F) \quad (2)$$

where d₀ and d₁ are the effective dose conversion factors corresponding to ²²²Rn gas and its progeny. The recommended values for these factors are: d₀= 0.33 μSv.a⁻¹ / Bq.m⁻³ y d₁= 80 μSv.a⁻¹ / Bq.m⁻³ [12].

2.2 Results

Table 1 shows the radon gas concentration average values measured for each analyzed province, detailing the number of dwellings in each case and the measurement methods used. In this table we found the total number of measurements performed since 1983 up today by the ARN laboratory.

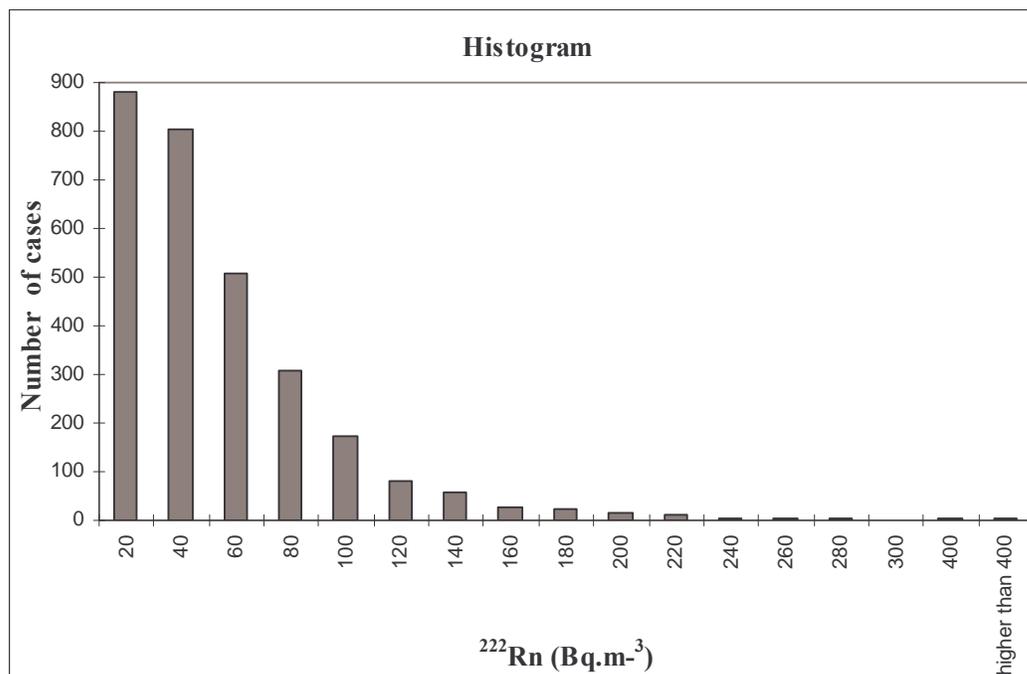
Table 1. Radon gas concentrations in Argentine dwellings

Province	Average radon concentration (Bq.m ⁻³)	Number of dwellings	Measurement method*
Mendoza	44,4	1198	1,2,3,4
Corrientes	48,0	109	1
Buenos Aires	33,2	464	1,2,3,4
Chaco	49,0	35	1
Santa Fe	31,0	61	1
San Luis	30,7	204	1
Chubut	70,0	357	1,3,4
Santiago del Estero	28,0	62	1
Rio Negro	35,2	73	1,4
Córdoba	46,4	321	1,4
Entre Ríos	79,4	17	4
<u>Argentina</u>	44,6	2901	

* 1 Makrofol, 2 Electrets, 3 Carbón activado, 4 CR-39

In Figure 3 it is shown the number of dwellings for every analyzed radon concentration range.

Figure 3. Histogram of ^{222}Rn concentration



On the other hand, table 2 shows the results of the equilibrium factor F and the radon gas concentration. The measurements shown in this table were performed with the method that uses Makrofold, and determines both radon concentration and the equilibrium factor simultaneously. It can be observed the average, maximum and minimum results measured from the C_{Rn} and F values. In addition, the corresponding effective doses (E) are estimated by means of equation 2.

Table 2: Results of ^{222}Rn concentration measurements, equilibrium factor (F) and effective doses (E).

Province	C_{Rn} (Bq/m ³)			F			E (mSv/y ¹)			n
	Average	Max.	Min.	Average	Max.	Min.	Average	Max.	Min.	
Buenos Aires	17,6	63,2	5,0	0,36	0,88	0,11	0,48	1,37	0,12	48
Mendoza	62,8	125,6	26,9	0,39	0,96	0,04	2,00	8,35	0,21	68
Córdoba	88,0	170,8	28,9	0,31	0,56	0,08	2,00	6,05	0,64	28
Rio Negro	34,0	81,0	9,9	0,34	0,63	0,08	0,84	2,07	0,07	10
Neuquén	33,6	59,3	24,4	0,25	0,50	0,04	0,67	2,01	0,22	8
Santa Cruz	24,3	36,2	17,3	0,47	0,80	0,18	0,88	1,17	0,53	6
Chubut	48,5	170	15,5	0,40	0,88	0,01	1,60	4,46	0,02	54
Promedio	49,3			0,37			1,44			222

2.3 Conclusions

Up to date, 2901 dwellings have been analyzed since 1983. Average radon gas concentrations in the different provinces vary from 28.0 Bq.m⁻³ to 79.4 Bq.m⁻³ being the arithmetic average from all the dwellings equal to 44.6 Bq.m⁻³.

There are very few values over 200 Bq.m⁻³ and only in one case a value over 400 Bq.m⁻³ was measured. In this case a follow-up is being performed to this dwelling. An important factor to take into account is that the two measurements performed in this dwelling were carried out in different seasons: the first measurement was performed during winter, when a lower ventilation may increase the radon concentration in closed spaces. The second measurement was performed in a warmer period of time. The dwelling was evaluated for the first time with a time integrated detector from May till August obtaining a radon average concentration of 626 Bq.m⁻³. The dwelling was evaluated again in November with activated charcoal detectors. Six detectors were placed and the average concentration was of 29.7 Bq.m⁻³ with a standard deviation of 12.5 Bq.m⁻³. This dwelling is being evaluated in this moments with long term detectors in order to measure the radon average concentration in the most unfavourable season of the year again. If the annual average exceeds 400 Bq.m⁻³ [13], it will be recommended to implement some remedial measures.

From the radon gas arithmetic mean estimated for our country, it was calculated the annual effective dose, resulting in 1.11 mSv.a⁻¹.

The average radon concentration value obtained from 222 sampled dwellings with the simultaneous measurement method resulted to be 49.3 Bq.m⁻³ and the average value of the equilibrium factor was of 0,37. The estimated annual effective dose was of 1.44 mSv. a⁻¹.

The measured equilibrium factor of 0.37 allows us to confirm the assumed equilibrium factor of 0.4.

It is observed that radon levels inside dwellings are within the acceptable values for the population, not justifying remedial actions, except in isolate cases that are still under study.

Acknowledgements

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Argentina Neutron Calibration Center

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Argentina Neutron Calibration Center

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Abstract. In Argentina, Nuclear Regulatory Authority (ARN in Spanish) has a neutron irradiation facility for personnel dosimetry research, dosimetry intercomparison studies, dosimetry performance test programs, training of health physics and safeguards personnel, and radiobiology research.

The facility, established to support regulatory activities, is located at Ezeiza Atomic Center far away from other facilities in order to avoid the use of shielding when it operates. The laboratory dimensions are: 14m long, 6m wide, and 4m high and the roof and walls are made of galvanized plate steel. The steel building is mounted on a 15cm thick concrete floor directly on the ground.

The laboratory consists of a neutron irradiator with the capability to contain two sources and a pneumatically operated carousel for selecting the sources, located in a shielded design below ground level, to fit in a 12" pipe 2 m deep. The exposure position is 2m over the floor.

The sources are Cf-252 (1.4510^8 n/s) and AmBe (1.13×10^7 n/s); the ²⁵²Cf source can be used either bare or surrounded by heavy water (D₂O). The facility has been completed with a calibration bank made of aluminium, 6m long, 1m wide, and 1.5m high and a track system for movement of phantoms or detectors.

The laboratory's scattering was studied for the ²⁵²Cf and ²⁵²Cf + D₂O sources, in free air irradiation conditions, with the MCNPX -V2.5f code, resulting in a contribution of less than 10%. This value was experimentally validated for a ²⁵²Cf source.

The facility's neutron reference fields were calibrated for ²⁵²Cf (bare and moderated with heavy water) by Los Alamos National Laboratory (LANL; USA) using a ROSPEC II transfer detector, which was calibrated by PTB in the thermal region and by National Institute of Standards and Technology (NIST) up to 20 MeV.

The Neutron Calibration Center (NCC) was established in the framework of a collaboration programme between ARN and DOE-ORNL.

KEYWORDS: *neutron laboratory; calibration.*

1. Introduction

Since 1995, the ARN has been working to establish a facility with the capability of conducting personnel dosimetry research, dosimetry intercomparison studies, dosimetry performance test programs, training of health physics and safeguards personnel, radiobiology research and verifications/calibrations neutron detection systems. The sources used, AmBe (over 15 years old) and ²⁵²Cf (2.4106 n/s – 1994) were quite exhausted. The irradiation system was simple and was manually operated. An important experience was developed in performing measurements applying radiation protection devices like Bonner Spheres System, ISO Hp phantom, remmeters, shadow cone and so on[1][2].

In 2003 in the framework of a collaboration programme between the ARN and DOE_ORNL, the conditions of the facility were improved. A neutron irradiator with remote control, a ²⁵²Cf source, and a moderation sphere were installed in 2005 and an AmBe source was bought in 2007 by the ARN. The facility was calibrated by LANL in the fields of ²⁵²Cf source bare and moderated with D₂O and by September of 2008 it will have been completed with the AmBe source [3].

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2. The facility

The facility, placed in an isolated area far from other facilities in order to reduce radiation protection concerns, is surrounded by a wire fence with controlled access by Security Forces. Figure 1 shows the laboratory's layout. Figure 2 shows the laboratory (irradiation room) and the control room.

Figure 1: Scheme of the facility with the laboratory and the control room.

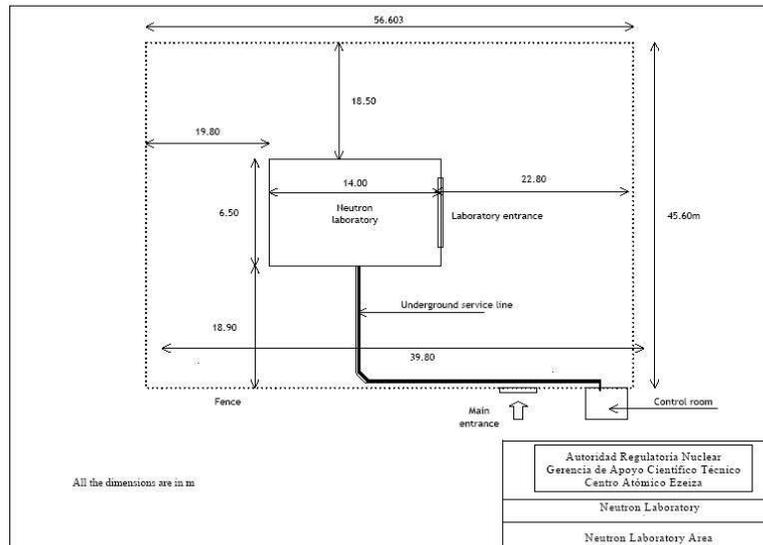


Figure 2: The Neutron Calibration Center (NCC): the laboratory and the control room



The radiation protection evaluation of the facility, considering the ^{252}Cf and AmBe sources stored underground and in irradiation position, was made with the MCNPX code [4]. The neutron spectrum was obtained from the ISO 8529 [5], and the ICRP-74 [6] conversion factors were applied in order to

obtain the effective dose distribution around the facility. The Effective Dose rate, E was calculated for ^{252}Cf , considering the maximum activity of a source and the gamma-neutron contributions at different positions: at 1m from the source, E is $0.78 \mu\text{Sv/s}$; at the fence, E is $100 \mu\text{Sv/year}$ (with an occupational factor of 0.061) and at the control room, E is $561 \mu\text{Sv/year}$ (occupational factor 1). The facility was determined to be in compliance with the ARN radiation protection regulations.

2.1. The Laboratory

The laboratory dimensions are 14m long, 6m wide, and 4m high and the roof and walls are made of galvanized plate steel. The steel building is mounted on a 15cm thick concrete floor directly on the ground.

The laboratory contains a neutron irradiator, two neutrons sources (^{252}Cf and AmBe), and a track system. The neutron irradiator provided by DOE – ORNL is an N40-BG-M-2 model manufactured by Hopewell Designs Inc, USA with the capability to hold two sources using a single exposure tube suitable for the larger AmBe source. The irradiator comprises a pneumatically-operated carousel which fits 2 meters down a 12-inch-diameter pipe for selecting sources and maintaining maximum shielding for the unselected sources which remain underground. As the irradiator is based on a pneumatic system, a compressor was also installed outside the laboratory. The exposure position, where the sources are raised to, is 2 meters above the floor. See Figure 3.

Figure 3: Neutron Irradiator



The system includes a remote controller with an electronic timer, a source selection, and a safety interlock interface to control a safety interlock system with an interlocked door and a scram switch.

The ARN and ORNL experts tested the system using a dummy source, and all the security systems were found to work properly.

The facility has been completed with a calibration bank made of aluminium, 6m long, 1m wide, and 1.5m high and a track system for movement of phantoms or detectors.

2.1.1. The sources

The neutron sources installed are ^{252}Cf and AmBe.

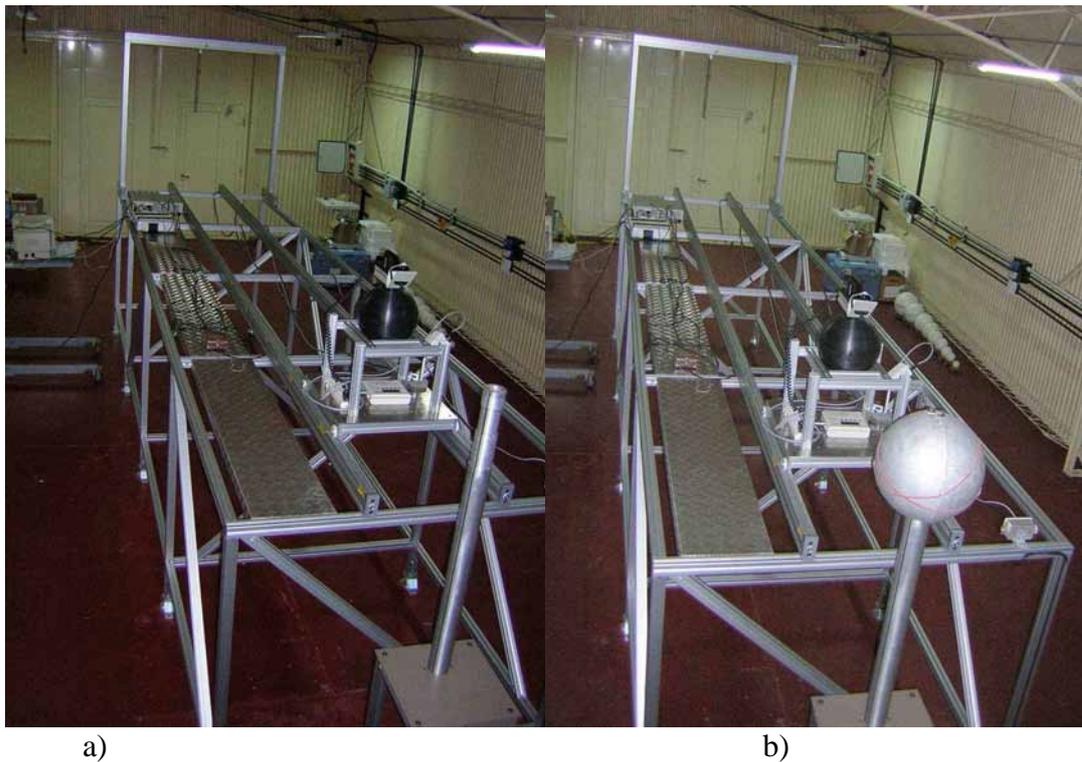
The ^{252}Cf source is housed in a type 100NS capsule, manufactured by Frontier Technology. The neutron emission rate, reported by the manufacturer, is traceable to the NIST and is stated as $2.8 \text{ E}+8$ n/s with a reference date of May 10th, 2005. The overall uncertainty associated with the neutron emission rate is estimated as 2% (1sigma). The calculated anisotropy factor is 1.19[3].

The ^{252}Cf source can be used bare or surrounded by a 30-cm D_2O -moderator sphere. The moderation sphere is made of stainless steel, 32.3 cm in diameter, with a removable 1mm cadmium cover. The sphere fits over the end of the calibration receiver so that the source, when in position, is in its centre.

The AmBe source is housed in a type X14 capsule, manufactured by QSA Global Inc. The neutron emission rate reported by the manufacturer is $1.13 \text{ E}+07$ n/s (22nd March 2007).

Figure 4 shows the laboratory with the calibration bank, the track system with a detector and the irradiator system with the guide for both sources a) bare and b) moderated.

Figure 4: Laboratory's Experimental arrangements for a) bare and b) moderated sources



2.2. The control room

The control room is a prefabricated cabin made of aluminium located outside the laboratory.

A movement alarm, video control, telephone line and radiation detectors were installed in the facility, all of them connected to the control room. All the control systems for radiation protection, physical security, and the neutron irradiator are in this room.

3. Characterizations of the Neutron Reference Fields of the Facility

The Neutron Reference Fields of ^{252}Cf bare and ^{252}Cf (D_2O , Cd) were characterized by Los Alamos National Laboratory (LANL) in September 2006. The neutron spectrum measurements were performed using the Rotating Neutron Spectrometer (ROSPEC) at 0.7 m, 1.5m, 3m and 5m from the source. The ROSPEC transfer detector was calibrated by PTB in the thermal region and by NIST up to 20 MeV. The experimental arrangement used is shown in Figure 5.

Figure 5 ROSPEC detector at 0.7 m from the $^{252}\text{Cf}(\text{D}_2\text{O}, \text{Cd})$ source



A theoretical study of the laboratory's backscatter for the ^{252}Cf and ^{252}Cf (D_2O , Cd) sources, in free air irradiation conditions, was made with the MCNPX (ref). The room scatter correction factor calculated by LANL and ARN is shown in Table 1. This factor which includes air attenuation and air in scatter as well as the albedo effect of the room is defined as the total dose rate divided by the dose rate without scattering contribution. Results showed a scattering contribution of less than 6% at 0.7 m and less than 60% at 5 m according to the characteristics of the laboratory described above, which minimize the scattering.

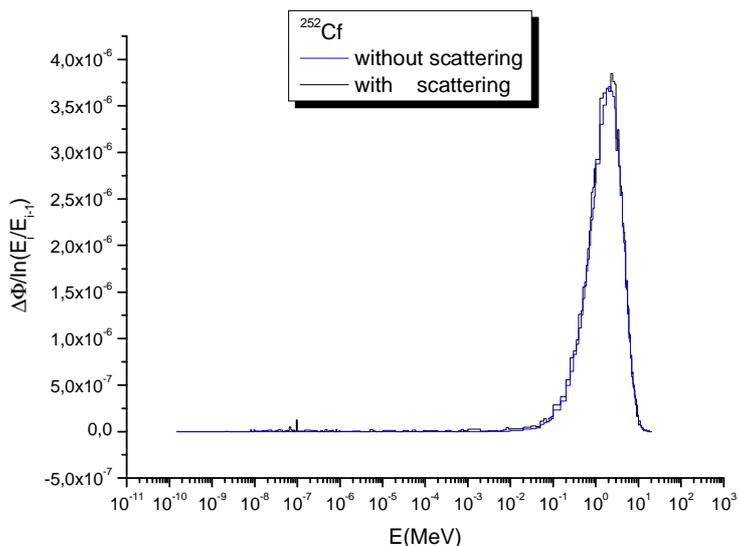
Table 1 Calculated Room Scatter Correction factors as a function of source- detector distance

Distance (m)	Calculated Scatter Factors	
	Bare ^{252}Cf	^{252}Cf (D_2O , Cd)
0.7	1.032	1.052
1.0	1.062 (ARN calculations)	1.068 (ARN calculations)
1.5	1.110	1.178
3.0	1.263	1.395
5.0	1.396	1.556

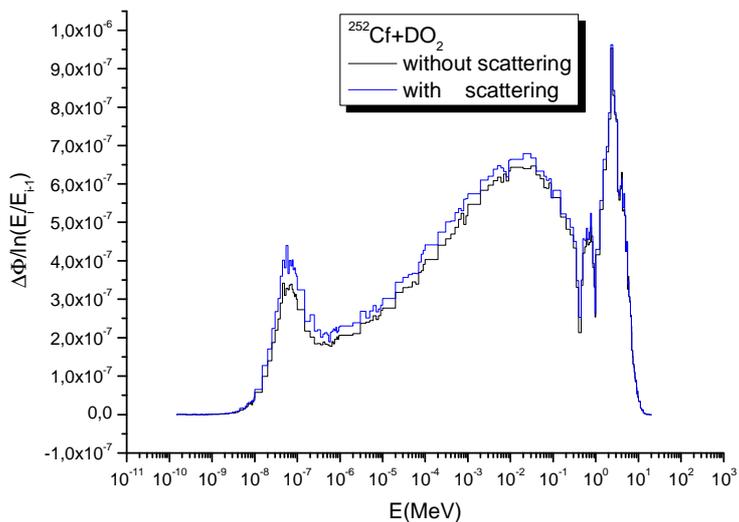
The Lethargic representation of the neutron spectra of ^{252}Cf , according to ISO 8529, and ^{252}Cf (D_2O) sources, at 1m from the sources, with and without room scattering are shown in Figure 6.

Figure 6 Lethargic representation of the neutron spectra of ^{252}Cf (a) and ^{252}Cf (D_2O)

(b), with and without room scattering



a)



b)

Comparing with other facilities[7], the NCC has a lower scattering factor at different positions 1m, (LANL: 1, 09 ; NCC: 1,065); 1.5 m, (LANL: 1,67 ; NCC: 1,11) ; 3 m (LANL:1, 48 ; NCC: 1,26); 5m (Savannah River National Laboratory: 2; NCC 1,396).

The spectral fluence rate in function of distance shows an increase in the thermal component for both sources, as it is indicated in Table 2.

Table 2: Fraction of the fluence's component as a function of distance.

Distance (m)	Fraction of the component of Fluence as function of the distance					
	²⁵² Cf			²⁵² Cf (D ₂ O, Cd)		
	Thermal	epi	fast	Thermal	epi	fast
0.7	1	1	1	1	1	1
1.5	4	1,3	0,97	2	1,02	0,95
3	10	2,1	0,91	3	1,01	0,83
5	17	2,8	0,88	5	1,06	0,76

The behaviour of the neutron spectra described is due to the global effect of the room scatter. It is necessary to take into account these spectra variations in the process of calibration thermal neutron detectors because of their energy response. Though it could not have an important effect in the value of the equivalent dose because the conversion factor fluence to dose for thermal region is lower than the corresponding to other energies by more than 50 [6]. Table 3 shows the Ambient Dose Equivalent Rate, H*(10), measured by ROSPEC and calculated by MCNPX, and the variation of the quotient between H*(10; r) and H*(10, 0, 7). Therefore the H*(10)'s variation is similar for both spectra.

Table 3: Ambient Dose Equivalent Rate and H*(10, r)/H*(10, 0.7) in a function of the distance.

Distance (m)	Bare ²⁵² Cf		²⁵² Cf (D ₂ O, Cd)	
	H*(10) Ambient Dose Equivalent Rate (μSv/h)	H*(10, r)/H*(10, 0.7)	H*(10) Ambient Dose Equivalent Rate (μSv/h)	H*(10, r)/H*(10, 0.7)
0.7	5.17 E+3	1,00	1.47 E+3	1
1.5	1.18 E+3	0,23	3.40 E+2	0,23
3.0	3.30 E+2	0,06	9.82 E+1	0,07
5.0	1.30 E+2	0,03	3.95 E+1	0,03

4. Programme Activities

National Program Intercomparison of personal dosimetry in mixed fields: ²⁵²Cf, ²⁵²Cf (D₂O, Cd) and ¹³⁷Cs has begun in the facility.

Two different types of neutron detectors (LB6411 (Berthold) and NM2B (Thermo Scientific)) were calibrated following the ISO-8529 methodology [8].

The BSS measurements in the laboratory are being conducted to compare them with the LANL results in order to agree upon BSS as a transfer instrument.

The characterization of the AmBe facility's field with a ROSPEC system/MCNPX code is scheduled to be performed during a visit from ORNL and LANL scientists in September 2008.

In 2009, an intercomparison of radiation protection neutron detectors is scheduled to be performed in Argentina.

The protocols required to calibrate detectors are being written in accordance with the ISO 8529[5].

Conclusions

Neutron Calibration Centre has been established in Argentina. The laboratory is ready to irradiate different devices and systems at three different fields: ^{252}Cf bare, ^{252}Cf moderated with D_2O , and AmBe.

The NCC is a low scatter facility, with scattering factors lower than others published by international laboratories. This objective of the DOE/ARN agreement has been fulfilled.

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The Fitness for Purpose of Analytical Methods Applied to Fluorimetric Uranium Determination in Water Matrix

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The fitness for purpose of analytical methods applied to Fluorimetric uranium determination in water matrix

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Abstract This paper describes the steps which should be followed by a laboratory in order to validate the fluorimetric method for natural uranium in water matrix.

The validation of an analytical method is a necessary requirement prior accreditation to a Standard ISO/IEC 17025 [1], of a non – normalized method.

Different analytical techniques differ in a sort of variables to be validated. Depending on the chemical process, measurement technique, matrix type, data fitting and measurement efficiency, a laboratory must set up experiments to verify reliability of data, through the application of several statistical tests and by participating in Quality Programs (QP) organized by reference laboratories such as the National Institute of Standards and Technology (NIST), National Physics Laboratory (NPL), or Environmental Measurements Laboratory (EML).

However, the participation in QP not only involves international reference laboratories, but also, the national ones which are able to prove proficiency to the Argentinean Accreditation Board.

The parameters that the ARN laboratory had to validate in the fluorimetric method to fit in accordance with Eurachem guide [2] and IUPAC [3] definition are:

Detection Limit, Quantification Limit, Precision, Intra laboratory Precision, Reproducibility Limit, Repeatability Limit, Linear Range and Robustness.

Assays to fit the above parameters were designed on the bases of statistics requirements, and a detailed data treatment is presented together with the respective tests in order to show the parameters validated.

As a final conclusion, the uranium determination by fluorimetry is a reliable method for direct measurement to meet radioprotection requirements in water matrix, within its linear range which is fixed every time a calibration is carried out at the beginning of the analysis.

The detection limit [4] (depending on blank standard deviation and slope) varies between 3 ug U and 5 ug U which yields minimum detectable concentrations (MDC) of 7 ug/l and 12 ugU/l respectively.

KEYWORDS: *Method Validation; uranium fluorimetric determination; Eurachem guide; analytical variables validation.*

1. Introduction

Reliable analytical methods are required for compliance with national regulations in all areas of analysis. It is accordingly internationally recognized that a laboratory must take appropriate measures to ensure that it is capable of providing and does provide data of the required quality. Such measures include:

- Using validated methods of analysis
- Using internal quality control procedures
- Participating in proficiency testing schemes
- Becoming accredited to an International Standard, normally ISO/IEC 17025

As analytical requirement, once the laboratory has faced an analytical problem, it should be defined the performance capabilities of the method to fit for the purpose.

Even though, the fluorimetry determination for natural uranium is applied since the early days of radiochemical methods, its application on molten pellets of sodium fluoride, still remains without normalization, therefore it is necessary to prove performance reliability through the analytical evidence of several parameters, such as Minimum Detectable Concentration; Linear Range, Robustness and Trueness.

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Precision, Intermediate Precision, Repeatability Limit, Reproducibility Limit, are parameters which permit to know about the closeness of results obtained for one operator or more than one, in the same laboratory in different times, (Precision and Intermediate Precision respectively). Then the respective limits are fixed:

$$\begin{aligned} \text{Repeatability Limit} &= 1.65 S_r && (95\% \text{ Confidence level}) \\ \text{Reproducibility Limit} &= 1.65 S_R \end{aligned}$$

2. Experimental

Water samples are taken to the uranium laboratory after filtered and acidified.

Three 400 μl aliquots are introduced in platinum dishes together with five blanks and a calibration curve (NIST traceable uranium standards). The whole set of dishes is evaporated until dryness prior to melt it in a high temperature oven during 5 minutes.

After 10 minutes cooling the dishes, at room temperature, the instrument is calibrated and the samples are measured in the same conditions within the working range of the calibration curve. Data are reported as μU (micrograms) per liter. The final calculation for the uranium concentration is a function of the slope, intercept and sample aliquot.

3. Experimental validation of the Fluorimetric method

3.1 Minimum detectable concentration (MDC)

In order to validate the MDC, we started defining the Detection Limit under the assumption that the standard deviation of the samples are independent of the signal and approximately constant respect to the standard deviation of the blank, which in terms of IUPAC model, is expressed as:

$$\text{MDC} = \frac{2L_c}{m * Val} * 10^3$$

Where

L_c : is the decision level defined as $3S_B$

S_B : standard deviation of the blank.

m : slope from the linear regression

Val : sample aliquot taken for analysis.

MDC: minimum detectable concentration.

According to the procedure presented to the Argentinean Accreditation Board (PP-FLUO-003), a calibration curve is obtained plotting instrument signal vs. increasing amounts of uranium certified standard. Five replicates of each point are measured, including the reagent blank.

Data are obtained by regression analysis considering the slope, intercept and residuals. From the blanks set, the mean blank is obtained and the standard deviation applied to MDC calculation.

The MDC as stated above is verified by measuring successive dilutions of the same uranium standard, until results become unreliable (table 1). Data obtained between the Decision Level and the Detection Level assess the MDC.

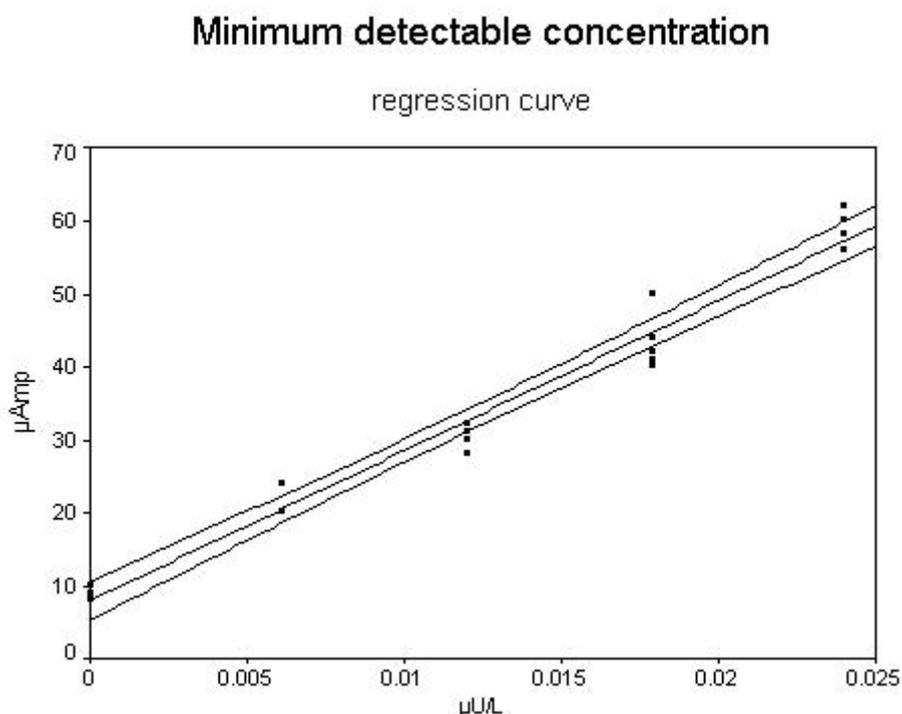
Table 1: Successive dilutions of the uranium certified standard measured as samples

SAMPLE	$\mu\text{U/l}$	Combined Uncertainty	Relative Uncertainty	Expanded
1	3.8	3.6	95.6	
2	6.3	3.2	50.7	
3	11.4	2.9	25.1	
4	16.8	2.8	17.0	
5	32.0	2.6	8.0	
6	46.4	4.7	10.1	
7	60.2	5.3	8.8	

From the Fig. 1, a possible lack of linearity next to the intercept (intercept is slightly shifted respect to the blank measurements) is observed.

The variability and high values of relative uncertainty [5]($K=2$) obtained for data close to the decision and detection levels (3.3 and 6.5 $\mu\text{U/l}$, respectively), are expectable and do not express reliability of the measurements at such low levels, therefore these data are used to asses MDC. Above the quantification level (higher values than 16.8 $\mu\text{U/l}$), the relative uncertainty tends to be minimum and constant.

Figure 1: Regression curve to assess MDC



3.2 Trueness

As defined in terms of bias, trueness is the difference between the average of a large series of measurements in repeatability conditions, and the value of either, a Certified Reference Material or a Test Sample provided by a reference laboratory. The results are assessed by application of a Student's test as follows:

$$t_{calc} = \frac{|V_R - V_L|}{\sqrt{\left(u_R^2 + \frac{S_L^2}{n}\right)}} \quad (1)$$

Where

- t_{calc} is the value for the calculated Student's test, according the effective degrees of freedom.
- V_R is the value of the Reference Material or Reference Activity from de Reference Laboratory
- V_L is the average value of the laboratory assessing trueness.
- u_R is the combined uncertainty of the Reference Material or Reference Activity
- $\frac{S_L}{n}$ is the average standard uncertainty of the laboratory assessing trueness.

The condition to consider that Trueness has been assessed is that : $t_{calc} \leq t_{tab}$

In order to perform this test, the sample N° 32 from the NRIP/06 Quality Program from the National Institute of Standards and Technology (NIST) was assayed. Results are shown in table 2

Table 2: Data obtained from NRIP/06 sample N° 32 on 18 replicates

SAMPLE REPLICATE 1 st DAY	MASS ACTIVITY µg/l	EXPANDED UNCERTAINTY (µg/l)
1	36.2	±1.2
2	36.2	±1.2
3	29.4	±1.2
4	31.7	±1.2
5	36.2	±1.2
6	38.4	±1.3
7	38.4	±1.3
8	30.5	±1.2
9	41.8	±1.4
SAMPLE REPLICATE 2 nd DAY		
1	34.5	±1.7
2	34.5	±1.7
3	36.6	±1.8
4	32.4	±1.7
SAMPLE REPLICATE 3 rd DAY		
1	31.0	±1.2
2	32.0	±1.3
3	35.0	±1.3
4	31.0	±1.2
5	35.0	±1.3
Average	34.26	S=3.37

The NIST reference sample assayed in this experiment was 32.54µg/l ±1.3% (K=2), and the combined standard uncertainty was 0.21 µg/l .

Then, the obtained value for t_{calc} (17 degrees of freedom) = 2.093 which is lower than the corresponding $t_{tab} = 2.120$

3.3 Robustness

The capacity of a method to stand the changes an operator introduces in it on the purpose, without changing its performance, is the method Robustness.

The most critical step in the fluorimetric method (measuring uranium in sodium fluoride pellets) involves the fluoride fusion at high temperature.

Variable amounts of sodium fluoride from 100 to 500 mg, were added to a set of 5 aliquots (three replicates each) of Reference Material from NIST, containing natural uranium and left to stand for 10 minutes. Assays were carried out changing the cooling time and the fluoride amounts before fusion.

Table 3: Time variation on 500 mg of sodium fluoride pellets.

TIME (min)	500 mg µg/l	UNCERT. µg/l
5	150.7	9.7
10	148.0	9.7
15	150.7	9.7
20	164.1	9.5
25	161.4	9.5

Table 4: Variable amounts of NaF

SERIE (mg NaF)	SAMPLE AVERAGE µg/l	UNCERT. µg/l
100	158.7	9.5
200	161.4	9.5
300	158.7	9.5
400	137.2	9.9
500	153.4	9.6

3.4 Linear Range

From the regression line obtained when calibrating the fluorometer, together with residuals and the correlation coefficient, a working range is fixed in the linear response of the instrument.

Five spiked blanks with reference material (NIST standard three replicates each), were prepared from the lowest to the highest concentration level, previously determined.

In order to asses the linear working range, measurements were performed on the samples as described above and results are shown in tables 5 and 6.

Table 5: Spike and experimental data in the linear range

SPIKE (µg)	EXPERIMENTAL	COMB.UNCERTAINTY (µg)
0.0061	0.0063	$9 \cdot 10^{-4}$
0.0120	0.0100	$7.0 \cdot 10^{-4}$
0.0179	0.0174	$8.0 \cdot 10^{-4}$
0.0240	0.0250	$1.3 \cdot 10^{-3}$

The assessment of linearity was performed applying the ANSI -N 42.22.-1995 (American National Standards Institute) on every concentration obtained.

$$|V_p - V_m| < 3 \sqrt{U_{mp}^2 + U_m^2}$$

$|V_p - V_m|$ is the bias between the standard and the concentration obtained.

U_{mp} : is the combined uncertainty of the standard.

U_m : is the combined uncertainty of the standard as sample.

Table 6: Data comparison and assessment of Linear Range

SAMPLE N°	ANSI -N 42.22.-1995
1	$2 \cdot 10^{-4} < 1.52 \cdot 10^{-2}$
2	$2 \cdot 10^{-3} < 1.51 \cdot 10^{-2}$
3	$5 \cdot 10^{-4} < 1.52 \cdot 10^{-2}$
4	$1.0 \cdot 10^{-3} < 1.55 \cdot 10^{-2}$

3.5 Participation in Intercomparison Quality Program

Since the fluorimetric method was incorporated to the Nuclear Regulatory Authority Quality Program, the laboratory has participated in several Interlaboratory Radiochemistry Test Programs organized by the Argentinean Reference Laboratory (CNEA) and the National Institute of Standards and Technology (NIST). In the table 7 the results are shown.

Table 7: Comparison of data obtained by the Fluorometry Laboratory NIST/2006 and CNEA /2007

REFERENCE INSTITUTION / YEAR	REFERENCE VALUE Bq/g		FLUORIMETRY LABORATORY- Bq/g	
NIST / 2006 (fluorimetric data were converted to Bq/g)	U-234	3.8790	U-234	4.0
	U-235	0.1854	U-235	0.19
	U-238	4.0	U-238	4.0
CNEA / 2007 data expressed in $\mu\text{U/g}$	MA	812.67	MA	862.0
	MB	370.05	MB	386.0
	MC	137.98	MC	153.0

4. Conclusion

Experiments to validate the parameters mentioned above, yield results statistically proven in either, comparing the Student "t" in the cases of Trueness and Linear Range, or by simple observation of measurement variability (Robustness and MDC).

Even though the MDC is depending on calibration curve (slope) and blank standard deviation, the MDCs obtained make it suitable the fluorimetric method for routine assays in the range of 10 $\mu\text{gU/l}$ up to 800 $\mu\text{gU/l}$.

However, at the extremes of the curve, some deviation from reference values is expectable, due to lack of linearity (top of the working linear range) and a lower response of the instrument close to the detection level.

The data obtained for validation of Robustness, show a little variability for samples left to cool longer than ten minutes, while no significative changes are observed increasing the amounts of NaF (there

was observed an unexpected result for 400 mg which is considered as random, due to a large variability in the repeatability of the measurement).

On the other hand, the results obtained in the Quality Programs show traceability to the standards sent from the reference laboratories.

As a final conclusion, the Fluorimetric method fits for the purpose of Radioprotection Monitoring of uranium in water, when applied to environmental samples under the above validated parameters.

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The fitness for purpose of analytical methods applied to Fluorimetric uranium determination in water matrix

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INTRODUCTION

The fluorimetric method for natural uranium in water matrix still remains without normalization. Therefore, it is necessary to validate the method if the laboratory wants to become accredited to an International Standard, such as ISO/IEC 17025. The parameters that the ARN Laboratory had to validate, according to the Eurachem guide, IUPAC definition and Argentinean Accreditation Board (OAA) are: Minimum Detectable Concentration, Trueness, Linear Range, Robustness, and Intra laboratory Precision.

EXPERIMENTAL PROCEDURE

Three 400 µl aliquots of water sample are introduced in platinum dishes together with five blanks and a calibration curve (NIST traceable uranium standards). The whole set of dishes is evaporated until dryness prior to melt it in a high temperature oven during 5 minutes.

After 10 minutes cooling the dishes, at room temperature, the instrument is calibrated and the samples are measured in the same conditions within the working range of the calibration curve. Data are reported as µgU (micrograms) per liter. The final calculation for the uranium concentration is a function of the slope, intercept and sample aliquot.

EXPERIMENTAL VALIDATION

Minimum detectable concentration (MDC)

The Minimum detectable concentration is verified by measuring successive dilutions of the same uranium standard, until results become unreliable. Data obtained between the Decision Level and the Detection Level assess the MDC.

Trueness

As defined in terms of bias, trueness is the difference between the average of a large series of measurements in repeatability conditions, and the value of either a Certified Reference Material or a Test Sample provided by a reference laboratory. The results are assessed by application of a Student's test as follows:

$$t_{calc} = \frac{|V_R - V_L|}{\sqrt{\left(\frac{u_R^2}{n} + \frac{S_L^2}{n}\right)}}$$

V_R is the value of the Reference Material or Reference Activity from the Reference Laboratory

V_L is the average value of the laboratory assessing trueness.

u_R is the combined uncertainty of the Reference Material or Reference Activity

$\frac{S_L}{n}$ is the average standard uncertainty of the laboratory

Participation in Intercomparison Quality Program

Since the fluorimetric method was incorporated to the Nuclear Regulatory Authority Quality Program, the laboratory has participated in several Interlaboratory Radiochemistry Test Programs organized by the Argentinean Reference Laboratory (CNEA) and the National Institute of Standards and Technology (NIST). In the table 1 the results are shown.

Robustness

The capacity of a method to stand the changes an operator introduces in it on purpose, without changing its performance, is the Robustness.

Linear Range

From the regression line obtained when calibrating the fluorometer, together with residuals and the correlation coefficient, a working range is fixed in the linear response of the instrument. The assessment of linearity was performed applying the ANSI N42.22-1995 (American National Standard Institute) on every concentration

$$|V_p - V_m| \leq 3 \sqrt{U_{mp}^2 + U_m^2}$$

$|V_p - V_m|$ is the bias between the standard and the concentration obtained for that standard.

U_{mp} is the combined uncertainty of the standard.

U_m is the combined uncertainty of the standard as a sample.

Table 1

REFERENCE INSTITUTION / YEAR	REFERENCE VALUE	FLUORIMETRY LABORATORY
NIST / 2006 Fluorimetric data converted to Bq/g	U-234 3.8790	U-234 4.0
	U-235 0.1854	U-235 0.19
	U-238 4.0	U-238 4.0
CNEA / 2007 Data expressed in µgU/l	Sample A 812.67	Sample A 862.0
	Sample B 370.05	Sample B 386.0
	Sample C 137.98	Sample C 153.0

CONCLUSION

Although only some of the parameters for the validation of the Fluorimetric method are shown, they are of major concern to fit this technique for the purpose, since acceptable MCD and a good performance in routine analysis are observed from the data obtained in evolution tests and quality programs organized by NIST and CNEA. To sum up, the Fluorimetric method fits for the purpose of Radioprotection Monitoring of uranium in water, when applied to environmental samples under the above validated parameters.

The Use of the Cynefin Framework in Emergency Management

Hernández, D.G.; Kunst, J.J.; Jordán, O.D. and Boutet, L.I.

Presentado en: 12th International Congress of the International Radiation Protection Association.
Buenos Aires, Argentina, 19-24 octubre 2008.
El presente trabajo presentó poster durante el evento.

The use of the Cynefin framework in emergency management

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Abstract

The Cynefin framework was developed by David J. Snowden as a tool for decision-making that has been used for knowledge management as well as in other applications. The Cynefin framework has four domains or spaces named known, knowable, complex and the chaos domain.

In the known domain a recognized relationship between cause and effect exists and the usual procedures work properly. In the knowable domain a cause and effect relationship also exists but it is incompletely known; if we have the resources and the required time to obtain information and additional knowledge it is possible to move to the known space. In the complex domain the cause and effect relationship only can be determined after the effect has happened and the experience obtained cannot be used to carry out new predictions. In the chaos domain a cause and effect relationship cannot be identified at all.

In some application of the Cynefin framework, there are no more desirable domains; however, this is not true during the handling of a major crisis, like a nuclear emergency. In this situation, the preferred Cynefin domains to work in are: the known domain, where the well established actions can be safely applied, as those that are practiced during exercises; The knowable domain, in which appropriate predictions can be made by obtaining information, i.e. monitoring, and using conventional models; the complex domain where environments should be created to make patterns more evident and stabilize the more advantageous, allowing to manage the situation properly or to move it to the knowable space. Finally, in the chaotic domain it is important to act quickly, perhaps in an authoritarian way, to reduce the disorder, evaluating the result and creating several patterns to move the situation into the complex domain; obviously, this is the less desirable domain, but in certain circumstances it is unavoidable.

This paper explores the application of the Cynefin framework to the preparation and response in case of nuclear emergencies, considering specially the actions to be taken to avoid that the situation move toward some of the not wanted spaces or, at least, to slow down that movement.

KEYWORDS: *Emergency planning, emergency preparation and response, nuclear emergency, Cynefin, decision making.*



THE USE OF THE CYNEFIN FRAMEWORK IN EMERGENCY MANAGEMENT

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The Cynefin framework was developed by David J. Snowden as a tool for decision-making that has been used for knowledge management as well as in other applications. The Cynefin framework has four domains or spaces named known, knowable, complex and the chaos domain.

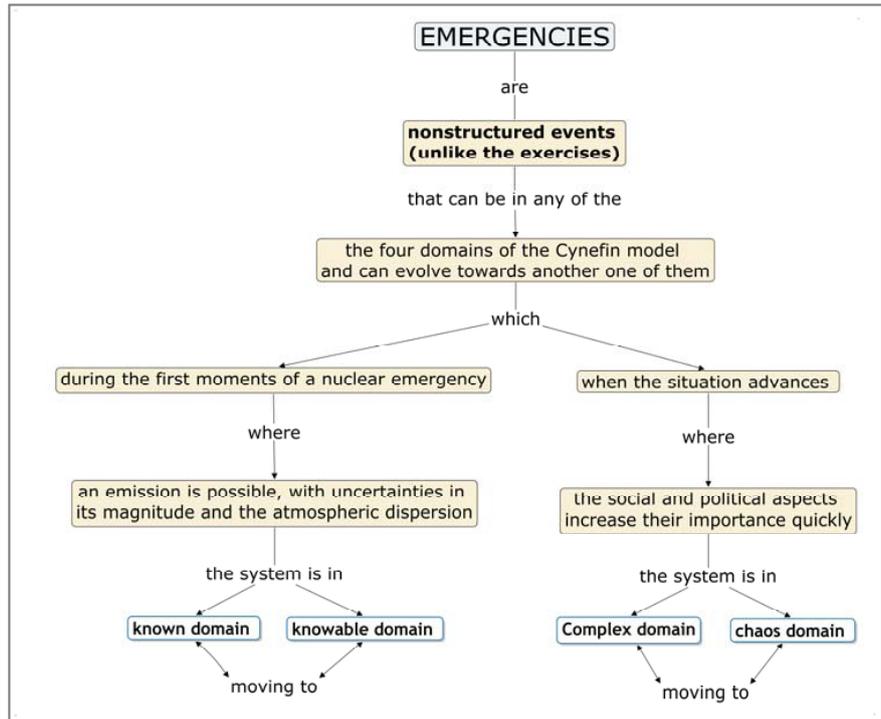
In the known domain a recognized relationship between cause and effect exists and the usual procedures work properly.

In the knowable domain a cause and effect relationship also exists but it is incompletely known; if we have the resources and the required time to obtain information and additional knowledge it is possible to move to the known space. In the complex domain the cause and effect relationship only can be determined after the effect has happened and the experience obtained cannot be used to carry out new predictions. In the chaos domain a cause and effect relationship cannot be identified at all.

The nuclear emergencies are complicated socioeconomic events, due to the strong interrelation of technological and social factors, which interact with the physical, political and economic environment. While in some application of the Cynefin framework, there are no more desirable domains, this is not true during the handling of a major crisis, like a nuclear emergency.

In that situation, the preferred Cynefin domains to work in are: the known domain, where the well established actions can be safely applied, as those that are practiced during exercises and the knowable domain, in which appropriate predictions can be made by obtaining information, i.e. monitoring, and using conventional models.

In the figure on the right we present an example of how the Cynefin framework can be used to characterize the different phases during the response to a nuclear emergency.



During the progress of a nuclear emergency it is essential to make suitable decisions and exercise the appropriate leadership style. The Cynefin framework is useful to determine in what domain is the system and, therefore, to define how to achieve the desired objective. The following table summarizes the most visible features of the domains of the Cynefin framework, the more appropriate kind of management or leadership and a summary of the main tasks of the leader.

Making decisions in different domains				
	Known	Knowable	Complex	Chaotic
Characteristics	Repeated patterns and consistent events. Clear and evident cause-effect relationship.	Cause-effect relationship is not evident but can be discovered. The diagnosis of experts is required.	Cause-effect relationship is not clear. Many decisions without time are due to take to think.	There is not a correct answer. Emergent patterns. Several ideas competing.
Kind of management – leadership	Management based on facts	Management based on facts.	Leadership based on patterns	Innovative leadership
Leader's tasks	Judge, categorize, respond. To be sure the suitable processes are available. To delegate	Judge, analyze, respond. Create experts panels. Pay attention to the conflicting information	Probe, judge, respond. Create atmospheres and experiments that allow patterns to emerge	Act, judge, respond. Looking for what works, instead of search for the correct answer

Conclusions

The Cynefin framework is a tool for making decisions that can be very useful to used in the preparedness and response for emergencies, allowing analyze the internal relationships of the response organization and the interaction of the organization with the socio-economic and political environmental that conditions and restricts their action. Based on the correct analysis of the situation it is possible to know what are the decisions best suited to stabilize the system or to move it to other domain.

The Handling of Nuclear Emergencies in Argentina

Hernández, D.G.; Jordán, O.D.; Kunst, J.J. and Bruno, H.A.

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El presente trabajo presentó poster durante el evento.

The handling of nuclear emergencies in Argentina

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Abstract

In 1998, the Executive signed the decree 1390, which defined the scope and the procedures corresponding to the Nuclear Activity Law. In this decree, the new functions of the Nuclear Regulatory Authority (ARN) are described, being the most important related to preparation and response for a nuclear emergency the following ones:

- ARN must provide protection from harmful effects of ionizing radiations under normal conditions and emergency situations.
- ARN must advise the Executive in case of radiological and nuclear emergencies.
- ARN shall establish the criteria for the emergency plans of the facilities and train the members of neighbor public to the facilities in case of nuclear emergencies.
- The emergency plans developed by local, provincial and national authorities must be approved by the ARN.
- ARN shall lead the actions within the area covered by the emergency plans of the facilities. Security Forces and the Representatives of Civil Institutions shall report the designated ARN officer.

The ARN recognized immediately the responsibility imposed by this law and, at the same time, the opportunity of improving the handling of emergencies through a centralized direction of the operations. Under this frame, ARN created the Radiological Emergencies Intervention System (SIER) with the goal of taking charge of the preparation and the handling of emergency situations.

From the beginning, the purpose of the SIER was to improve the preparation and response to nuclear emergencies in a regular form, bearing in mind the cultural and socioeconomic situation of the country, as well as the local peculiarities. The first step to achieve such a target was to gain the confidence of other organizations included in the response on the ARN technical and operational aptitude to lead the actions inside the emergency area and, later, to establish the pertinent arrangements.

The strategy chosen by ARN to respond to nuclear emergencies consists in establishing an expert team and a decision-making team at its Headquarters, with the responsibility of conducting the whole emergency. At the same time, the ARN shall deploy a team to the emergency area, led by the Operative Head of Nuclear Emergencies Response- "Jefe Operativo de Emergencias Nucleares (JOEN)" - who is in charge of the response at the local level (up to about ten kilometers). The JOEN is also responsible for the coordination of other organizations included in the response at the local level.

This paper presents the organization of the preparation and response in case of nuclear emergencies that was developed in Argentina, considering the particular responsibilities assigned to ARN by the current legislation. In addition, three essential topics in this working scheme - the training of the members of the public, the coordination of the different organizations and the development of knowledge and skills in ARN relevant areas- are developed.

KEYWORDS: *Emergencies, Emergency Preparedness, Emergency Response.*



THE HANDLING OF NUCLEAR EMERGENCIES IN ARGENTINA

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Background

In 1998, the Executive signed the decree 1390, which defined the scope and the procedures corresponding to the Nuclear Activity Law. In this decree, the new functions of the Nuclear Regulatory Authority (ARN) are described, being the most important related to nuclear emergency the following ones:

- ARN must provide protection from harmful effects of ionizing radiations under normal conditions and emergency situations.
- ARN shall establish the criteria for the emergency plans of the facilities
- ARN shall train the members of neighbor public to the facilities in case of nuclear emergencies.
- The emergency plans developed by local, provincial and national authorities must be approved by the ARN.
- ARN shall lead the actions within the area covered by the emergency plans of the facilities. Security Forces and the Representatives of Civil Institutions shall report the designated ARN officer.

Under this frame, ARN created the Nuclear Emergencies Intervention System (SIEN) with the goal of taking charge of the preparation and the handling of emergency situations.

Purpose of the SIEN

The purpose of the SIEN was to improve the preparation and response to nuclear emergencies in a regular form, bearing in mind the cultural and socioeconomic situation of the country, as well as the local peculiarities.

The strategy chosen by ARN

- To gain the confidence of other organizations included in the response on the ARN technical and operational aptitude to lead the actions and, specially, to gain the confidence of the public.
- Establishing an expert team and a decision-making team at its Headquarters, with the responsibility of conducting the whole emergency.
- Deploy a team to the emergency area, led by the Operative Head of Nuclear Emergencies Response (JOEN), who is in charge of the response at the local level (up to about ten kilometers). The JOEN is also responsible for the coordination of other organizations included in the response at the local level.
- To implement a set of automatic protective actions, that must be completely prepared as soon as the possibility of a nuclear emergency is detected and implemented before the emission occurs.

The automatic protective actions that are implemented in the urgent protective action zone (10 km of the power station in the wind direction) are: control of accesses, distribution of iodine tablets, sheltering and, within 3 km in all the directions, evacuation of the population. Also the population is kept informed of the progress of the emergency and about the implementation of the protection actions.

The ARN considers emergency exercises a fundamental tool to fulfill the stated objectives and responsibilities, in several forms.

- They allow practicing the tasks to realize during a real emergency.
- They confirm the confidence of other organizations in their own operational capabilities and strengthen the confidence in the ARN as the leader organization.
- They offer the best places to spread information about the real risks and to train the population.
- They allow determining the opportunities of improvement in the emergency plans and their application.
- They allow testing new technologies available.



Distribution of iodine tablets, access control, ingestion of simulated iodine tablets and preparing for sheltering during the 2008 Exercise at Embalse NPP.

The tasks of providing information to the population during the preparation of emergency exercises are considered one of the most important, since they improve the behavior of the population. These tasks are carried out in particular at schools (first and second levels, students from six to seventeen years old), since students give the best response and provide wide spreading of the information.



Providing information to the public. Students from primary school and general population.



An aerial view of Embalse NPP from the unmanaged aerial vehicle "LIPAN" designed and operated by the Argentine Army, during the 2008 Exercise at Embalse NPP. The LIPAN can be seen in the insert.

Conclusions

After more than 50 nuclear emergency exercises carried out at Atucha and Embalse NPP 1973, we verified the correctness of the strategy chosen by ARN because it had allowed fulfilling the proposed objectives and legal requirements, allowing the required flexibility and the fluid integration of several organizations under the direction of the ARN.

Preparation for the Members of the Public in Case of a Nuclear Emergency in Argentina

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Preparation for the members of the public in case of a nuclear emergency in Argentina

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Abstract. The Nuclear Law (Law N° 24804, year 1997) establishes that the Nuclear Regulatory Authority (ARN) has the duties of population's informing and preparation for the case of nuclear accidents. Knowing the foresee involvement of the public in a nuclear emergency, this preparation is carried out in the framework of contingency plans approved by the Nuclear Regulatory Authority. The population's preparation is a key task to arrive to the objective of minimizing the consequences of a nuclear incident. The ARN has defined the necessary countermeasures for the first hours and as the local population is involved, they must be trained in order to ensure the better level of efficacy. An important part in the preparation is the diffusion of information in the schools so the students know what they should do in case they receive an alert from the nuclear emergency command. The presentations at schools are divided by the differences in the levels of understanding, considering the ages of the students. These tasks are carried out once a year during the exercise preparation and it has a main objective: to diffuse the primordial protection actions. Similarly diffusion activities on protection actions were carried out with other sectors of the general public, with talk shows, discussions and explanation of doubts generated during the open meetings invited by the local civil defense, the nuclear power plant and the ARN. This work presents the details of tasks carried out with the educational community, the civil organizations the population and the conclusions obtained during these activities.

KEYWORDS: emergency planning, emergency preparation and response, nuclear emergency, public information.

1. Introduction

The work carried out by the Nuclear Regulatory Authority in connection with accidents that involve radiation and radioactive material, which could affect workers, people from the public and the environment, is aimed primarily at establishing a policy of prevention.

An ongoing task of preparation is conducted with civil organizations that work with people, especially with the Civil Defense, local police, firefighters, public and private schools, hospitals and medical services and community organizations. All this is for the preparation of the public in the event of a nuclear accident.

In addition, with the agreement of the educational system authorities of the place, the students of different schools are trained to carry the protect actions message to their homes.

The experience has shown that, involving population and its organizations the earliest possible, in decision aiding processes will generally favours the mutual understanding of the situation.

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Although it will probably increase the timing of the process, involving stakeholders will generally allow a better cooperation between all participants and lead to more acceptable and robust decisions.

This task involves the public, civil institutions (private, national, provincial and municipal) and Security Forces within the radius where it is necessary to implement the municipal emergency plan and to start the primary protection actions (10 km. around a nuclear power plant during the first hours after an accidental situation begins).

This work presents some details of tasks carried out with the educational community, the civil organizations and the population and the conclusions obtained during these activities.

The aim is to avoid or minimize the initial consequences of a significant accidental release of radioactive material of a nuclear power plant, from the people, environment and property, complementing other actions from the Civil Defense to protect themselves, promoting the organization and training of publics, civil institutions and security forces in the vicinity of a nuclear power plant in a "framework" of prevention.

2. Legal Framework

The Nuclear Law (Law N° 24804, year 1997) establishes that the Nuclear Regulatory Authority (ARN) has the duties of population's informing and preparation for the case of nuclear accidents. Knowing the foresee involvement of the public in a nuclear emergency, this preparation is carried out in the framework of contingency plans approved by the Nuclear Regulatory Authority.

The Nuclear Regulatory Authority sets the criteria for drawing such emergency plans that basically aim at putting in place the functional elements required to respond in case of a nuclear emergency

The law also establishes that the law enforcement and the civil institutions within the area affected by an emergency must consider such procedures and they shall respond to an official of the Nuclear Regulatory Authority.

3. Regulatory Framework

The Operation License for NPPs in Argentina establishes that they shall have an emergency plan approved by ARN and it must be verified at least once a year. This plan must include the on- and off-site aspects of a nuclear emergency.

The facility emergency plans should be consistent with the municipal, provincial and national emergency plan. They must shape a response system to ensure coordination of the various organizations involved led by the Nuclear Regulatory Authority.

4. Prevention Criteria

Keeping in mind this annual obligation, mechanisms for ensuring members of the public participation in the exercises are in place. The population's preparation is a key task to arrive to the objective of minimizing the consequences of a nuclear incident.

It is known that the level of public protection during a nuclear emergency should be the best under the existing circumstances, so the preparation of the public to understand a nuclear emergency (as the emergency is developed and what the public should do in case of emergency is produced), is faced to minimize the doses or risks received.

5. Protection Action

The Nuclear Regulatory Authority has defined the necessary countermeasures for the first hours and as the local population is involved, they must be trained in order to ensure the better level of efficacy.

The main radiological countermeasures consists in an early evacuation until 3 km, access control, and sheltering and iodine prophylaxis in the potential affected areas, initially established as 10 km around the Nuclear Power Plant.

For the early evacuation, around the 3 km, the National Gendarmerie has a detailed and updated list of people living in the area, and is the institution to perform this task.

Before a radioactive material release begins, stable iodine tablets will be delivered to all residents in the affected area originally defined by the wind direction. The distribution of the tablets will be realized by the National Gendarmerie in schools, public places and houses. After tablets distributing, the emergency staff will give instructions to residents through radio or trough of sirens, speakers, about when ingest it.

Residents should expect the announcement on local radio stations to swallow the pill. The local radio stations will send messages produced by the organization's management response, and will inform the public about the events that occur in the facility. The role of local radio is very important because, basically, indicate the moment of ingesting the stable iodine tablet.

In the case of sheltering, people must stay inside the houses, public buildings, etc, with doors and windows closed. This provides protection for the incorporation of radioactive materials transported by air to breathe, by direct radiation produced by them and deposited on surfaces.

The access control means to control the entry and exit to the affected area by the nuclear emergency at strategic points, to facilitate the implementation of protect actions. This task is carried out by the local police and civil defense.

6. Diffusion Activities

Diffusion activities are oriented primarily to explain a nuclear accident, and what actions are in place to protect local populations and to minimize the radiological risks. The Nuclear Regulatory Authority together with the facility operator and the local civil defense have implemented training programmes for people and organizations involved in the area covered by the emergency plan. The training is oriented towards the effective implementation of protective actions and is carried out by the Nuclear Regulatory Authority.

An important part in the preparation of the population is the diffusion of information in the schools so the students know what they should do in case they receive or listen the alert from the nuclear emergency command. The presentations at schools are divided by the differences in the levels of understanding, considering the ages of the students. For this purpose, four levels are defined.

They are: Level 1 for children from 6 to 9 years old, Level 2 for children of 9 to 12 years old, Level 3 for teenager of 12 at 15 years old and Level 4 for teenager of 16 to 18 years old. These tasks are carried out once a year during the exercise preparation and it has the main objective of diffuse the primordial protection actions.

Similarly diffusion activities on protection actions were carried out with other sectors of the general public, with talk shows, discussions and explanation of doubts generated during the open

meetings invited by the local civil defence, the nuclear power plant and the Nuclear Regulatory Authority.

The contents of these activities explain how the electricity is generated through a nuclear power plant, citing the advantages and technological difficulties involved. Also this activity explains how a nuclear accident is developed, which is the impact on the public and why these protect actions minimize the impact. Basically we send a clarified message to the people: that an accident at a nuclear plant is not a nuclear explosion, as many people believe. All this task of preparing the public is done in a common language adapted to the participants, taking care not to include very detailed explanations and words somebody would expect from a non-technical language understandable for anyone who is not in theme, in order to not confuse the public, and to make protective actions to minimize the risks be understood.

7. Conclusion

As a result of this activity, where civil organizations municipal, provincial and national, security forces, armed forces, operators of nuclear power plants, media and public diffusion, participate together with the Nuclear Regulatory Authority and people living in the area (with emphasis on Students of all levels in schools and neighborhood organizations), people involved in planning this becomes a "resistant community" to address other common emergencies, such as those due to weather effects (severe storms, floods, etc. .), accidents conventional industry (toxic chemicals, etc.).

Meteorology and Dispersion Forecast in Nuclear Emergency in Argentina

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Guichandut, M.E. and Chiappesoni, H.

Meteorology and Dispersion Forecast in Nuclear Emergency in Argentina

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Abstract. The “Nuclear Regulatory Authority (NRA) (ARN in Spanish)” and the “National Meteorological Office (NMO) (SMN in Spanish)” of Argentine has been working together on the improvement of both meteorological forecasting and dispersion prediction.

In the pre-release phase of a nuclear emergency, it is very important to know the wind direction and the forecast of it, to establish the area, around the installation, where the emergency state is declared and to foresee the modification of this area. Information is also needed about deterministic effects, to begin the evacuation.

At this time, meteorological forecast of wind direction and speed, and the real time meteorological information is available in the nuclear power plant (NPP) and in the Nuclear Emergency Control Centre at the ARN headquarter, together with the short-range dose calculation provided by our dispersion code, SEDA.

By means of the SEDA code, we can estimate the optimum place to measure the radioactive material concentration in air, needed do to reduce evaluation uncertainties due, among others, to poor knowledge of the source term. the SEDA code allows considering atmospheric condition, and the need to reduced doses of the measuring team in charge of the measurements.

For the evaluation in the medium range, we participate in the project IXP, which provides four hours and about 50 kilometres forecast. In the long-range movement of air borne radioactivity, the World Meteorological Organization (WMO), whose contact point in Argentina is the SMN, can assist us. We have developed together, with the SMN, a detailed procedure to request assistance from the WMO.

In this work, we describe the combined tasks that were carried out with the SMN to define the procedures and the concepts for their application during a real emergency. The results of an application exercise carried out in 2006 are also described.

KEYWORDS: *Nuclear emergency, meteorological forecast, atmospheric dispersion*

1. Introduction

Nuclear Regulatory Authority (ARN) and the National Meteorological Office (SMN) of Argentina, has been working together on the improvement of both subjects meteorological forecasting and dispersion prediction.

This cooperation was born from necessity to improve the response during an emergency situation, to a potential or actual release of radioactive material

The time after the recognition of situation that could lead to a release of radioactive material to the environment (pre-release phase) should be used by emergency organization to declare the emergency and establish the area of emergency, to tune the actual situation with preplanned countermeasure. In this context is necessary to know the actual and forecast wind direction, to determine the general emergency zone where we will take preplanned countermeasure, evacuation, sheltering, and iodine pill distribution. (Fig.1)

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Fig.1 Emergency zone



2. System and Instrumentation

2.1 Meteorology during pre-release phase

In this phase is necessary to know de direction and velocity of wind direction, the cloud cover, cloud high and the insolation, to determine the stability class, and high of the plume to introduce it in radiological consequence codes SEDA 4.78, (Sistema de Evaluación de Dosis en Accidentes). It is based on a short range Gaussian plume model (10 to 20 Km) of atmospheric dispersion (1) developed by the Nuclear Regulatory Authority to determine projected doses for bone marrow dose, internal organ dose, due to immersion in the plume, inhalation and deposition for 52 nuclides to evaluate the necessity of extend the replanted protection actions

2.2 Meteorological Instrumentation near NPP and Atomic Centre.

Nuclear Power Plant and Nuclear Atomic Centres has meteorological instrumentation to be used during normal operation or emergency situations. Today we are developing methods to determine Pasquill-Gifford stability class by temperature gradient, sigma theta methods measurement of solar radiation.

The meteorological information is transmitted to ARN Meteorological Information Centre in Ezeiza Atomic Centre, and it will be use via internet in the ARN Emergency Control Centre

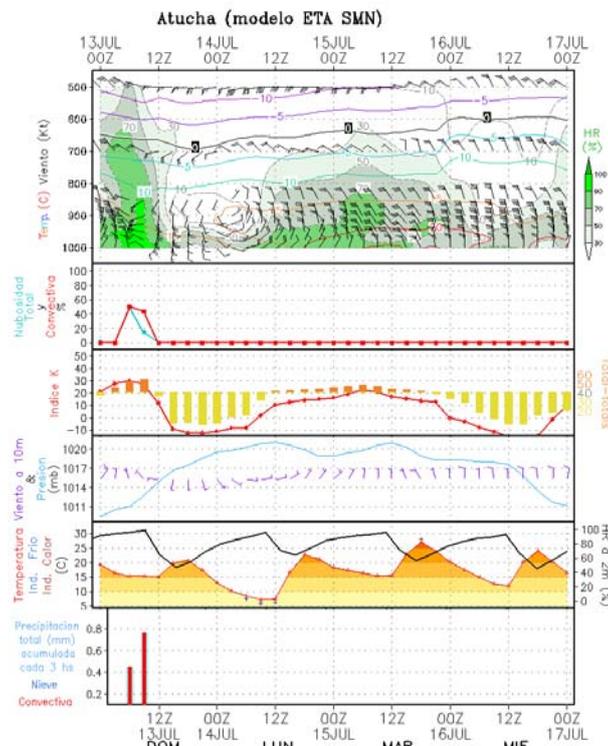
2.3 Meteorological and Dispersion Forecast

2.3.1 Meteorological Forecast

During the pre release phase or in the released phase is necessary to know the forecast of wind direction and wind intensity to determine Pasquill-Guifford stability class and the possibility of rain to the emergency zone or elsewhere. NWS (SMN in Spanish) is in this case which provides forecasts of this variables and weather condition given that it have trained staff and use the support of ETA SMN high resolution numerical forecast model (25 km horizontal), which is run two time a day (0 and 12

UTC). The ETA SMN model could be used to forecast 120 hs. in advance, at 3 hours intervals, in 38 pressure level high, and the variables are temperature, specific humidity, horizontal wind, pressure of surface, turbulent kinetic energy, water and ice in the cloud.

Fig 2 Atucha NPP Meteogram



2.3.2 WMO-IAEA Atmospheric dispersion Asístanse

The WMS designed a response process to nuclear or radiological emergency that is in accordance with the suggested procedures by WMO. The WMO coordinate the assistance by Regional Specialized Meteorological Centre (RSMC) in Atmospheric dispersion, which give assistance in atmospheric dispersion with advanced air dispersion and deposition models. You can access the modelling results via internet to a website with an authorized permission give by WMO to authorized centres, which are accepted by RSMC. [1]

The Meteorological Centre of South America has been assigned to the RSMC of Washington (EEUU) and Montréal (Canada)

2.3.3 “International Exchange Project (IXP)” Atmospheric Dispersion assistance

IXP web interface allows the user to do plume modelling for basic release scenarios, and the predictions are for four hours.

IXP users can

1. Quickly enter a simplified description of a radiological atmospheric release
2. Send the information electronically to NARAC (National Atmospheric Release Advisory Center) for processing
3. Receive an initial atmospheric dispersion and dose prediction based on NARAC’s 3-D models
4. Easily generate a standardized consequence report

The report describes the health effect consequences associate with a hypothetical release to the atmosphere from a radiological source. This is an initial NARAC product. Prediction should be confirmed and refined using measurements.

Products

- Total Deposition
 - Ground Exposure Dose Rate
 - Total Effective Dose Equivalent
5. Easily download the results as a shape file for use in a geographical information system (GIS)
 6. Receive refined dispersion and dose prediction (produced by NARAC operation scientists with appropriate authorization [2])

To obtain an IXP account, an individual must be nominated by their country's National Competent Authority, and approved by the US DOE NNSA Office of International Emergency Management and Cooperation (NA-46).

3 Exercises

3.1 Routine Exercise

During nuclear emergency exercises at nuclear power plant of Argentina, which take place once a year, the communication between Emergency Control Centre of ARN (Nuclear Regulatory Authority) and SMN (National Meteorological Service) is exercised asking for the running of Meteorological Dispersion Model.

Also during nuclear emergency exercises, the ARN, run their own code for near field atmospheric dispersion SEDA and LLNL Hotspot. And used IXP (International Exchange Project from NARAC)

3.2 Real Time Exercises.

During July 13 of 2006 take place a real time exercise that consist

- a) Sending of meteorological data from Nuclear Power Plant Embalse (Embalse, Cordoba Province, Argentina), to the ARN Emergency Control Centre
- b) The ARN Emergency Control Centre calls on meteorological dispersion advice with Request for RSMC Support by Delegated Authority form.
- c) The ARN Emergency Control Centre runs near field Code Seda.
- d) The SMN send the form and received the information from RSMCs Washington, Montréal, Melbourne
- e) The SMN send the information to Emergency Control Centre, Maps with a Joint Statement which about deposition patterns

The atmospheric dispersion data were available 1 hour after of the exercise beginning. This exercise was considered successful, in all their parts. In the future we could receive shape files, to integrate this information with our geographical information system (GIS)

4. Conclusion

We have implemented the necessary tools for meteorological forecast and atmospheric dispersion for near field, mesoscale, and far field with help of IAEA and NNSA.

We need to develop the interaction between atmospheric dispersion code and measurement.

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Continuous Improvement of the Regulatory Framework for the Control of Medical Exposure

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Continuous improvement of the regulatory framework for the control of medical exposure

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Abstract

Background: One of the key elements to guide the improvement of the regulatory control is the availability of a self-assessment tool for regulatory performance. Although there is general guidance on self-assessment for regulators and users (IAEA), there is a need for more specific advice on how to address challenges and difficulties faced by regulatory bodies, when regulating radiation protection of patients. Examples of these challenges are the need for regulatory initiatives, in cooperation with health and education authorities, professional bodies and equipment suppliers, and to put in place necessary elements that are beyond responsibility of individual users of radiation, to enable them compliance with safety standards. **Purpose:** within the programme of the Ibero American Forum of Nuclear and Radiation Safety Regulatory Organizations, a project to develop such a self assessment tool for the regulatory control of medical exposure has been designed. **Method:** national experiences in transposing and enforcing the international radiation safety standards, as to how the requirements are included in national regulations are reviewed. Further, difficulties to the implementation of safety requirements are analyzed and a self assessment approach and possible regulatory solutions are presented. **Results and discussion:** In this study the following documents are being produced: 1) Transposition of international requirements into national regulations in the six countries of the Forum, 2) difficulties to implement and enforce the requirements, 3) guidance on self assessment of regulatory framework for medical exposure, 4) suggested contribution to the revision of international radiation safety standards.

Keywords: regulatory, medical exposure, self assessment, continuous improvement.

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1. Introduction

Medical uses of radiation bring unquestionable benefits to diagnosis and treatment of disease. These benefits increase continuously, due to the fast development of new technologies and techniques on one

hand and to the need to make the benefits available to developing countries. Every day more than ten millions of diagnostic radiology exams, hundred thousand nuclear medicine exams and more than thousand patients are treated with radiation [1]. As a result of this use, medical use of radiation is the largest exposure from man-made radiation sources [2]. Radiological risks range from trivial to serious depending on the type of application.

General radiography still accounts for about two thirds of all x-ray imaging. Wide variations in dose, up to two orders of magnitude, for many procedures resulted in increasing attention to determine how to ensure patients receive no more dose than necessary. Surveys of patient doses and image quality, and reference levels are a means of reducing doses while keeping diagnostic confidence. Countries using this approach have seen substantial decreases in these doses.

Conventional films are being replaced by digital techniques. These techniques have the potential for reducing dose, given their higher sensitivity. However, in the short-term, doses are likely to increase since higher patient dose usually means improved image quality; even though this improved quality is not always necessary for diagnosis. Also, with digital radiology, it is very easy to obtain (and delete) images, so there is a tendency to obtain more images than necessary.

The use of relatively high-dose procedures such as computed tomography (doses higher than conventional radiography), has been steadily increasing over the years, as new technology allows faster image acquisition and improved image quality. This is associated to a substantial increase in population exposure. In addition, in some countries there is a trend for promoting CT for early detection of diseases in asymptomatic patients, without being referred by a physician, and oblivious of the substantial dose involved.

Interventional procedures using x rays provide the highest exposure to individual patients among all imaging procedures with the possibility for radiation injuries. It has been reported that the severe injuries can be avoided or the severity could be largely reduced.

New complex radiotherapy techniques have been developed, such as beam intensity modulation, image guided radiotherapy, breathing gated radiotherapy, tomotherapy, radiosurgery, new treatment planning systems, virtual simulation and “all inclusive” electronic information systems, which pose new challenges from the safety view point. Severe, unintended exposures have been reported and continue to occur and preventive measures are of high priority.

Regulatory work has to be done in this context and radiation protection principles and requirements are to be applied in all these areas. There are a series of international documents [3,4,5] dealing with the elements of a national infrastructure for protection and safety, and the subjects to be addressed and questions to be asked to evaluate quality and effectiveness of the regulatory body in the various areas of responsibility (regulate, authorize, inspect, request corrective measures and, if necessary, enforce them). However, there is a need for reference documents with practical solutions and answer to the questions asked.

In the preamble of the International Basic Safety Standards for the Protection against Ionizing Radiation and for the Safety of Radiation Sources (the BSS) [3] the need was recognized to “provide facilities and services that are essential for radiation protection and safety, but are beyond the capabilities required of the legal persons who are authorized to conduct practices”. To meet certain regulatory requirements, there is a need for qualified and certified medical and paramedical staff, which is not available in a number of countries, for arrangements for generic justification of practices and techniques, calibration at a standards laboratory, quality audits, diagnostic reference levels, dissemination of lessons from accidental exposures, and assessment of safety of new technologies and equipment.

These are all elements which cannot be generated by the users individually not by the Regulatory Body alone. It needs cooperation, agreements, and arrangements with other institutions

(educational, health, labour and customs) and professional bodies in order to create and maintain the infrastructure. To make compliance with regulatory requirements possible, the regulatory body needs to take a proactive approach. On the other hand, the provision of such elements at the national level “does not detract from the ultimate responsibility for radiation protection and safety borne by the legal persons authorized to conduct the practices”.

2. Objectives

This project has a double overall objective: 1) to present options and practical solutions to enable regulatory programmes to become effective, making it possible for users to meet regulatory requirements on radiation protection and safety, and 2) to facilitate regulatory body’s self-assessment of performance on the control of medical exposure, thus contributing to its continuous improvement.

3. Method

The approach taken in this project consists of four actions: 1) the regulatory programmes of the FORO’s countries regarding medical exposures were analyzed. The analysis included the legal frameworks, the scope of the responsibilities of the regulatory bodies involved in patient radiation protection in each country, the regulatory infrastructures and the regulatory frameworks considering the transposition and implementation of the requirements of the BSS for medical exposure; 2) as a result of this analysis, the most common difficulties and obstacles in the national programmes were identified; based on the findings from 1) and 2), a document was written to assist regulatory bodies on self-assessment, with options and solutions to overcome the difficulties and obstacles found in step two, so as to enable users of radiation in medicine to comply with the safety standards; finally, 4) with these findings, a set of suggestions to improve the international safety requirements of the BSS are drawn as a contribution to the BSS revision process, which is currently ongoing.

4. Results

The implementation of the project has produced the following outcomes, in terms of difficulties, weaknesses of the regulatory programmes and the national infrastructure, and solutions towards compliance with BSS requirements and equivalent national regulatory requirements

4.1. Regulatory system

4.1.1. Division of regulatory responsibilities

Regulatory responsibility for different practices or different aspects of radiation protection and safety may be divided between different authorities. Regardless of the division of regulatory responsibilities, the preamble of the BSS states that the government must ensure that all aspects are covered [3]. However, division of responsibilities may lead to somewhat artificial separation. For example one regulatory body may be responsible for radioactive materials and another one for radiation generators, with the result that one and the same facility with both types of devices is regulated by two or more different bodies, responsible for regulating radiation protection and safety. Another example is that one authority is responsible for therapeutic applications and another one for radiology equipment for imaging, which brings new difficulties in radiotherapy, where increasingly extensive use of imaging equipment is made

In the absence of a systematic coordination mechanism among these regulatory bodies, there may be inconsistencies in regulatory requirements, in approaches to licensing, in inspection frequencies, in enforcement policy, in priorities for patient safety and even different safety culture in the regulatory bodies as such, with the appearance of contradictory opinions. All this may increase the administrative burden of users and not contribute to safety and in summary, may weaken the governmental credibility.

Understanding and coordination among regulatory bodies requires commitment and cooperative agreements at the highest level, in order to make coordination effective. The subjects of these agreements should cover the following aspects:

- Harmonization of the regulatory framework in order to minimize gaps and overlaps
- Coordination of the authorization and inspection process in order to reduce redundant administrative burden to users. A regulatory flow diagram of the authorizations and other regulatory actions would contribute to this coordination.
- Exchange of information in order to collect scattered information, provide feedback to inspection programmes, alerts on non-compliance and to have access to sensitive information related to accidental exposure of patients
- Joint activities to disseminate and promote safety culture, which should contribute to a solid national regulatory system.
- Joint regulatory policies, including enforcement, in order to strengthen national programmes on radiation protection of patients.

4.1.2. *Graded approach*

The current BSS [3] [paragraph 2.33] establishes that “technical requirements shall be applied when appropriate and to an extent commensurate with the magnitude and likelihood of the exposures expected from the practice or source”, which could be called a graded approach. This requirement is not specifically developed in the appendix on medical exposure, although more specific requirements to establish a graded approach in therapeutic, interventional and diagnostic procedures would be beneficial to use resources in an effective manner. A categorization of sources according to risk has also been published [6]. Subjects for a graded approach would be

- Degree of detail required for safety assessment for authorization, inspection frequency and follow-up on deficiencies
- Training programme for the staff of the various practices
- Degree of presence and involvement of qualified experts such as medical physicists, with full dedication in therapeutic applications and frequent availability in interventional procedures
- Degree of requirements on clinical dosimetry, including individual dose determination for therapy with unsealed sources
- Efforts to avoid accidental exposure and radiation injuries
- Degree of details for informed consent of patients. For high-dose therapeutic and interventional procedures, the informed consent should be in writing.

A graded approach to the system of authorization and inspection helps optimize regulatory efforts with a prioritized scheme of control. With a graded approach, facilities with higher risk, such as radiotherapy with new technologies, may benefit from more complex and proactive safety assessments, such as probabilistic methods and risk matrix approaches, also adapted and used in other projects of the FORO.

4.1.3 *Enforcement issues*

Social pressure makes difficult to restrict operation of a medical activity. This social scrutiny asks for optimized, balanced and sound enforcement actions, based on the principle of “do more good than harm”. A credible and respected regulatory body should base enforcement actions on the following pillars:

- Solid generic safety assessments of the radiological medical practices and techniques

- Specific inspection programmes designed on a risk based graded approach
- Competent and well trained staff with specific knowledge on the medical activities and hospital environment
- Institutional support at the highest level for the technical decisions

4.2. Closing gaps in national infrastructures

4.2.1. *Responsibilities for medical exposure and availability of qualified staff*

In many parts of the world, especially in developing countries, the inexistence of qualified professionals who are crucial to patient safety remains an unresolved issue. Moreover, compliance with regulatory requirements on medical exposure depends on the availability of qualified and certified radiation oncologists, nuclear medicine specialists, technologists and medical physicists.

In particular, medical physicists have responsibilities to meet the requirements on calibration of radiation beams and sources, clinical dosimetry, and quality assurance. They are unavailable in many countries and there are two reasons for this situation: 1) there is no programme of education and on-the-job training established in many countries, and it may be unfeasible to maintain such programme at a national level, where only a small number of professionals is required; 2) sending professionals abroad for education and on-the-job training often results in losing them because they prefer to stay in the country where they have been trained, particularly if the profession is not even formally recognized as a health profession in their own countries.

It is the responsibility of governments to be aware of these difficulties and to make provisions for a system for education, to have in place a process of accreditation, to formally recognize the staff as health professionals and to put in place a programme to keep in the country the staff that is crucial to safety. Although the approval of training on radiation protection falls under the responsibility of regulatory bodies, the education of staff in their own profession does not. Nevertheless, without availability of safety critical professionals in the country, regulatory requirements can not be met.

For these reasons, the regulatory body needs a proactive approach to work together with educational institutions, the health authority and professional societies in order to formally establish means to educate safety critical professionals and certify and recognize them as health professionals. The regulatory body should play a major role in this process, since it is the only authority with the information and knowledge of the safety implications of not having the necessary qualified experts and of the magnitude of the problem. In countries in which the number of hospitals, for example radiotherapy departments, is too small to maintain an educational programme cooperation with other countries and a joint strategy may be necessary.

4.2.2. *Justification of practices*

Justification of medical practices and techniques requires a judgment of benefits and risks. The regulatory body for radiation protection is normally not responsible for making judgments on the benefits of diagnosis and treatment with radiation. A sensible approach to generic justification of a given practice or technique and for a health screening programmes on asymptomatic patients at national or local level would be to obtain approval by the health authority in consultation with professional societies and regulatory body, the latter being in charge of evaluating or providing judgment on the risks of the application of radiation.

Since the primary benefit of medical exposure is for individual patients, decisions for individual exposure is up to the patient and his/her doctor, the latter bearing the responsibility to do more good than harm [7, 8]. However, professional bodies should provide guidance on appropriateness

criteria for each type of examination and treatment. This guidance can be used as reference by doctors prescribing and performing the procedure. There is recently a trend to promoting certain screening diagnostic medical exposures by some hospitals and clinics, leaving the decision for patients to make. This approach is called “self-referral”. An example of it is a total body scan for health screening, a technique with relatively high exposure.

These evaluations, judgments, guidance material and approaches to deal with controversial issues such as self-referral, are issues requiring participation of health authorities, regulatory body, medical professional societies and patient societies. They can only be dealt with in cooperation and with well coordinated programmes. The regulatory body, responsible to enforce compliance with requirements on patient protection needs to be proactive to develop institutional relations, permanent forums, ad-hoc task groups and other arrangements to make compliance with the regulatory requirements on justification possible.

4.2.3. *Optimization of protection*

4.2.3.1. Protocols for calibration and quality assurance

The first international conference on patient protection in 2001 [9] and the resulting Action Plan [10] concluded that requirements for medical exposure should be performance oriented and that details should be covered by guides, which can be adapted to the rapid evolution of equipment and techniques.

There are a number of protocols in different parts of the world and the language and abbreviations as well as the nomenclature in the formulas may not be entirely consistent. A misinterpretation and incorrect use of one of these variables and parameters may cause serious accidental exposure, especially in radiotherapy. Each professional will tend to use a different protocol, depending on the country he/she has more contact with. It is cumbersome for the regulatory body to monitor safety in these circumstances

The regulatory body should be proactive and work with professional bodies, encouraging them to adopt one of the protocols, preferable an international one, because international protocols are the result of an extensive harmonization work of the existing bibliography worldwide [11,12,13]. In this way, the adopted protocol becomes the recognized one by the health authority and the regulatory body and professional societies as a national consensus. This provides a unified approach, a common terminology and parameters, which facilitates regulatory control, assistance among hospitals, training, external verification and double checks and quality audits as needed.

4.2.3.2. Metrology arrangements for calibration of dosimetry instruments

The BSS requires that dosimetry equipment be traceable to a standards dosimetry laboratory. A valid calibration certificate is, therefore, an essential element to safety. To maintain traceability over time, dosimetry equipment needs to be recalibrated periodically. The process of sending, calibrating and returning the dosimetry equipment should be completed in a short period of time, because hospitals need the equipment permanently available for regular measurements.

There are a number of issues that may cause delay, such as administration, budget, transport, waiting time in the laboratory, return. This is aggravated in countries where there is no national laboratory, and equipment needs to be sent abroad for calibration. Experience has shown that equipment may be retained in customs for months or even import fees may be requested in both countries.

It is necessary to put in place mechanisms to expedite the process and facilitate compliance with regulatory requirements on traceability. These elements and arrangements go beyond the capabilities of individual users. There is here another opportunity for the regulatory body to be proactive and work together with the standards dosimetry laboratory, (for example organizing timely campaigns), or

with the customs authority to facilitate export and import. It may be also necessary to have cooperative agreements with another country with national laboratory. Contacts between the regulatory bodies and agreements between countries may make this process possible.

4.2.3.3. Reference levels and mechanisms for their establishment

The (BSS) [1] requires the use of guidance [reference] levels by medical practitioners in the process of optimization. These levels provide guidance on what is achievable with current good practice, and are to be applied with flexibility to allow higher exposures if these are indicated by sound clinical judgment.

In countries, such as the UK, where reference levels have been developed and applied for about 12 different types of examinations, a survey showed dose reduction of 30% and another survey performed about five years later, showed an additional reduction of 20% [14]. This significant reduction is primarily due to increase in awareness of radiation doses and optimisation of the radiology practices throughout the country.

It is well recognized that there are some circumstances in which overzealous reductions in patient dose can have a deleterious effect on image quality, which would be detrimental for patients. Methods for monitoring image quality are, therefore, required [15, 16, 17,18, 19, 21]. This ensures that guidance [reference] levels are associated with acceptable images and that any actions on patient doses resulting from applying guidance [reference] levels do not result in a loss of diagnostic confidence. This is why the BSS also requires that optimization is done taking into account norms of acceptable image quality established by appropriate professional bodies and relevant guidance [reference] levels for medical exposure. The BSS further establishes that guidance [reference] levels are to be established by relevant professional bodies in consultation with the regulatory [body].

The Action Plan on the Radiological Protection of Patients [10] indicates that guidance [reference] levels should be established locally. Differences in training, sensitivity of image receptors and quality control may cause differences, which can not be suddenly changed by imposing guidance levels from other countries. In order for medical practitioners to be able to use guidance [reference] levels they have to be established by professional bodies in consultation with the regulatory body. Establishing guidance levels is complex, and requires coordinated effort, and possibly a step-by-step approach, from a pilot exercise to a broader survey. Thus, the Regulatory Body should be proactive to get this process running and to monitor the use. Collaborative effort and agreements with the health authorities and professional bodies is essential. A review of the different approaches to reference levels worldwide is provided in [21].

4.2.3.4. Arrangements for acceptance of equipment

The BSS requires users to ensure that equipment conforms to international IEC and ISO or equivalent national standards, which also evolve with time and place increased requirements. Manufacturers have subsidiary responsibilities for the compliance with the BSS. They should contribute to facilitate generic safety assessments of specific components or complete equipment systems [5]. The generic assessment would be documented and such documentation of approved equipment is available in several of the industrialized countries. It is important for the user to ensure, before placing an order, that the equipment to be ordered “type approved” or carries a certificate of compliance with the IEC or nationally recognized equivalent standards in the country of use.

There are a number of difficulties associated with compliance with these BSS requirements, since old equipment may not comply with current IEC standards. Second hand equipment is some times donated, purchased, imported may not meet the standards either.

An inventory of equipment and its state of compliance should be made and an action plan with appropriate timing and deadlines may be needed. Again, a cooperative effort between the regulatory body, the health authority, manufacturers, and professional societies is required to develop

acceptability criteria and testing and a proactive approach of the regulatory body to move forward is crucial.

4.2.3.5. Arrangements for external audits

The BSS requires licensees to put in place a comprehensive programme of quality assurance for medical exposure, including the verification of relevant physical and clinical aspects. One of the problems to implement a complete programme is the difficulty of having external audits. The scope of the audit and the specialists to take part of an external audit is still a question of debate. The next question is the availability of audit results to the regulatory body, taking into account that external audits contain clinical aspects of medical practice, not necessarily related to regulatory requirements and that interference with medical practice may occur.

A suitable strategy to solve this problem would consist of cooperative agreements among the regulatory body, the health authority and professional societies on external audits regarding the presentation of reports. Subject of these agreements would be the application of risk-based graded approach and a method of reporting, in such a way that aspects related to regulatory requirements be filed separately. The risk-based graded approach would help select the relevant audit report subjects, starting by the risk of an accidental exposure (radiotherapy) or radiation injuries (interventional procedures) or diagnostic examination with higher exposure, for example computed tomography, or examination with a large impact to collective dose to the population. In this way these aspects can be provided to the regulatory body upon request without interference with other clinical aspects of the medical practice.

4.2.3.6. Database for disseminating lessons to avoid accidental exposure

BSS requirements include the investigation of accidental medical exposure, the adoption of corrective measures to prevent them in future and the report to the regulatory body, the patient and his/her doctor. Knowledge of information on a previous accidental exposure can help prevent a similar event in the future.

Not only can the knowledge of accidental exposure help prevent other events. Also near misses without consequences can provide important lessons to others. Sharing this knowledge is essential but obtaining this information is not straightforward. The staff of a hospital involved in a near miss may not be willing to share it because of fear that it may trigger actions by the hospital administrators or by the regulatory body. Reassuring confidentiality in this type of reporting is essential. There are some well known anonym reporting systems such as Radiation Oncology Safety Information System (ROSIS) collecting events with no consequences.

The Regulatory Body should take the initiative in designing a strategy, together with the health authority and professional societies for radiotherapy, interventional radiology, interventional cardiology and medical physics to stimulate communication and dissemination of information to prevent new events in the future.

4.3. Summary of suggested cooperation with other authorities and professional bodies

In the sections 4.2 and 4.3 a number of recommendations are given to the regulatory body to address gaps in regulatory and radiation protection infrastructures by means of cooperative approaches with other regulatory bodies, health authorities, professional bodies, manufacturers, customs authorities and patient societies. In addition cooperation with other countries may be necessary.

These recommendations are summarized in the following Table

Institutional agreement	Aspects to be covered	Main institutions directly involved
a) Fostering education and training for human resources on patient protection	<ul style="list-style-type: none"> ● Availability and recognition of the specialties for the professionals involved in patient protection ● Existence of a mechanism for their accreditation. ● Training of referring physicians prescribing medical exposure. ● Inclusion of radiation protection into medical curricula at degree level ● Training of specialists not directly involved in using radiation. ● Continuous professional education of medical and non medical staff 	Regulatory Body (RB), Health Authority (HA), Education Authority (EA), academic institutions and professional certifying agencies.
b) Fostering i of the various medical practices using radiation	<p>Inclusion of qualified experts in the health staffing</p> <p>Sufficient staff with responsibilities for protection and safety concomitant with work load.</p>	Regulatory Body (RB), Health Authority (HA), Education Authority (AE), Labor authority (LA), professional bodies for physicians, physicists and technologists
c) Facilitating mechanisms for justification of medical radiological practices	<ul style="list-style-type: none"> ● Development of standards ● Development or adoption of medical protocols for justification of medical exposure for diagnostic and interventional procedures. Special emphasis on sensitive groups (pediatric, pregnant and chronic patients) ● Development of guidance related to justification, such as appropriateness criteria for medical exposure ● Safe introduction of new technologies 	Regulatory Body (RB), Health Authority (HA), medical professional bodies
d). Fostering the establishment of diagnostic reference levels (DRL)	<ul style="list-style-type: none"> ● Inventory of equipment for diagnostic radiology and interventional procedures using X rays ● Dose surveys ● Specific training on establishing and using DRL ● National database on patient dose data 	Regulatory Body (RB), Health Authority (HA), medical professional bodies of radiology, interventional procedures using X rays, standards dosimetry laboratories, Universities
e) Fostering calibration and quality assurance programmes for the medical radiological	<ul style="list-style-type: none"> ● Metrology infrastructure to facilitate traceability in the calibration of radiation sources and beams used for medical 	Regulatory Body (RB), Health Authority (HA), medical professional bodies, standards dosimetry laboratories, suppliers of medical

Institutional agreement	Aspects to be covered	Main institutions directly involved
practices	exposure and for dosimetry equipment <ul style="list-style-type: none"> ● Adoption of protocols for calibration of sources and beams for clinical use ● Infrastructure for monitoring patient exposure ● Quality criteria for diagnostic images ● Protocols for recording physical and clinical data. ● Existence of preventive maintenance programmes ● External audits 	equipment and radioactive sources
f) Fostering the establishment of acceptability criteria for medical radiological equipment	<ul style="list-style-type: none"> ● Criteria for obsolete and old equipment (extension of “life”) ● Criteria for decayed radioactive sources for clinical use ● Import and manufacture of equipment in compliance with internationally recognized standards (IEC) or equivalent national standards ● Recommendations to use equipment incorporating devices for determination of patient doses 	Regulatory Body (RB), Health Authority (HA), medical professional bodies, suppliers of medical equipment and radioactive sources
g) Establishment of constraints for the release of patients treated with radionuclides or permanent implants	<ul style="list-style-type: none"> ● Environmental surveys and of socio-cultural conditions ● Writing recommendations (I-131) 	Regulatory Body (RB), medical professional bodies, for nuclear medicine and radiotherapy
h) Strengthening the metrological infrastructure and strategy for calibrations	<ul style="list-style-type: none"> ● Arrangements for calibration abroad when there is no laboratory in the country ● Agreements and arrangements with customs authority for export and return of dosimetry equipment abroad ● Arrangements for calibration campaigns and optimization of resources to satisfy the need 	Regulatory Body (RB), Health Authority (HA), Customs Authority (CA), standards dosimetry laboratory, medical physics society
i) Fostering emergency preparedness to deal with radiological accidents involving patients	<ul style="list-style-type: none"> ● Training in the early diagnosis of radiation injuries for the network of assistance in the case of an emergency 	Regulatory Body (RB), Health Authority (HA), Other institutions involved in emergency response
j) Establishment of a national database of radiological incidents and accidents	<ul style="list-style-type: none"> ● Awareness raising ● Assurance of confidentiality of the information ● Computer and information 	Regulatory Body (RB), Health Authority (HA), medical professional bodies for radiotherapy and interventional procedures using X rays

Institutional agreement	Aspects to be covered	Main institutions directly involved
	technology support ● Use of information for prevention, quality improvement, training and continuous professional development	

4.4. Summary of dissemination of safety culture related information

An essential element of the regulatory programme for the control of medical exposure is a policy of proactive dissemination of information on safety and regulatory issues to improve radiological protection of patients. The dissemination strategy should identify audiences, messages and means to disseminate. Information is one of the most powerful tools for the regulator in order to foster and consolidate a safety culture. Budgetary provisions for the dissemination programme are essential. The considerable expertise required for the sophistication of radiation protection in medical practice, and its importance, given the fact that it is the largest man-made exposure from man-made radiation sources, should be reflected in the budget. Funds should be also sufficient to cover all sectors involved in radiological medical practices.

Audience	Main information	Method
Patients	Existence of the regulatory framework Basic and specific simple information	web, leaflets, informative campaigns
Users of radiation, medical and paramedical professionals	Legislation, regulations, guidance and authorization conditions Forms and questionnaires Informative notes from the manufacturers on safety issues General information on protection of patients Information on incidents and accidental exposures	Web, mailing, workshops, bulletins, institutional publications, meetings
Health authority	Responsibilities and implications of the radiological protection of patients	Forums, round tables, institutional communications, working meetings
Professional bodies	Role of professional societies in the regulatory process for the radiological protection of patients. General and specific technical information on the radiological protection of patients	Workshops, forums, mailing, working meetings, round tables, congresses, round tables, institutional communications, Web
Policy-makers	Needs for the radiological protection of patients and consequences in case of lack of it or its degradation Need to provide a balanced budget to sustain programmes on the radiological	Institutional communications and presentations Public dissemination

Audience	Main information	Method
	protection of patients Need for policy of cooperation with other national organizations (mainly the health authority, education authorities and academia centers)	
Decision-makers (directors, managers, hospital administrators)	Compliance with regulations on the radiological protection of patients, Need for support to users of radiation Their responsibilities to foster and implement safety culture	Meetings, workshops, Web
Manufacturers and administrators	Need to comply IEC standards or national equivalent standards Conditions of acceptability of equipment	Institutional communications, meetings, workshops, forums, Web
Maintenance services	Awareness raising of the implications of maintenance errors with medical equipment on the radiological protection of patients	Institutional communications, meetings, workshops, forums, Web
Media	Public information on the radiological protection of patients, relevant information in cases of radiological accidents involving patients	Institutional communications, interviews, articles, Web,

4.5. Self-assessment of regulatory framework and continuous improvements

The IAEA radiation safety appraisal and support to Member States [22,23] refers to the need for the regulatory body to hold institutional relations. This criterion can be considered accomplished when coordination and cooperation at national level have been established and maintained with other authorities, intervening organizations, customs, law enforcement, professional societies, universities and technical services, as appropriate. This includes the formal and clear definition of respective responsibilities and functions. At the appraisal the following question is asked:

Does the regulatory body advise and co-operate with other relevant national authorities in relation to the implementation of the regulatory programme ... ?

Criteria from the IAEA Thematic Safety Area on Protection in Medical Exposure of the IAEA [23] refer to the need for cooperation with professional societies, suppliers, although there is no mention of the health authority.

In the work done under this project institutional relations and cooperative agreements have been addressed in full detail with about 43 evaluation questions on the subjects described in 4.2 and 4.3 of this report. This is the first time that institutional relations are considered in such detail. These kind of detailed questionnaires have been prepared for various aspects of the regulatory programme such as the regulatory framework, selection and training of the staff of the regulatory body and the authorization and inspection system for radiological medical practices. The goal is to assist the regulatory bodies in a self assessment process designed to analyze their ability to design and implement strategies for making regulatory programmes in the medical area efficient and effective. The proposed evaluation system is aimed at finding out not only “what” to do but “how” to do it.

4.6. Contribution to the review process of the International Basic Safety Standards

Important remark: these suggestions are the result of the work done by the authors and do not necessarily represent the position of the FORO's countries or of the FORO as such.

1. The current International Basic Safety Standards establish requirements for registrants and licensees only, not for governments or for regulatory bodies, because the BSS are based on the presumption stated in its preamble, that governments already discharge their responsibilities for protection and safety, by having legislation in place, a regulatory authority established and a national infrastructure for radiation protection and safety. The revised Standards should address governmental responsibilities and obligations of the regulatory body by means of explicit requirements.
2. Requirements should be included on the need for the regulatory body to reach cooperative agreements with health authorities, professional bodies, manufacturers, customs authorities and patient societies, in order to make provisions, which go beyond the capability of individual users, thus enabling them to comply with regulatory requirements. In addition, cooperation with other countries may be necessary for calibration of dosimetry equipment abroad when necessary.
3. Generic justification for practices and techniques requires the judgment on their benefits by health authorities and the risks considerations by the regulatory authority. Professional bodies may provide complementary guidance on appropriateness criteria for different diagnostic and therapeutic procedures. Requirements on governmental arrangements for justification are needed in the revised BSS, as well as clear assignment of responsibilities to referring physicians and medical radiological practitioners for individual medical exposures.
4. A graded approach should be incorporated in the revised version of the BSS, in order to allocate more resources where the risk is more important, ranging from high-dose therapeutic applications and interventional procedures down to low-dose simple diagnostic examination. Requirements for a graded approach are needed for both regulatory bodies and users of radiation. The tasks directly affected by the level of risk are listed in 4.1.2.
5. There should be requirements on governments to provide for educational programmes for the professionals needed for medical exposure. In particular, these programmes should address accreditation and recognition as health professionals for certain paramedical staff with substantial responsibilities for patient protection, such as medical physicists.
6. The current BSS takes for granted that national infrastructures provide for essential services, including standards dosimetry laboratory. In the revised BSS, requirements for governments should be included to provide this service or to make arrangements with customs authorities and with laboratories in other countries, in order to enable hospitals to maintain calibration of their radiation sources and beams used for medical exposure traceable to a standard dosimetry laboratory, through calibrated dosimetry equipment.
7. Requirements should be placed on the health authority and the regulatory bodies to work together with relevant professional societies for the establishment of diagnostic reference levels to be used as part of optimization of protection.
8. Requirements on the regulatory body should be included to foster safety culture in the country by establishing and maintaining a system of safety relevant information for parties affected by its decisions, medical and paramedical staff, the public, patients, manufacturers and other interested parties.
9. A requirement on the regulatory body should be included in the revised BSS to foster the establishment, in cooperation with the health authority, relevant professional societies, manufacturers and users, of a data base and mechanism for collection and timely dissemination of

information of lessons from incidents, near misses and accidental exposures, particularly from interventional procedures and therapeutic use of radiation.

10. A requirement on the regulatory body should be included, to establish, in consultation with health authority and relevant professional societies, criteria for acceptability of equipment, its useful life and obsolescence of medical radiological equipment.
11. A requirement on the regulatory body should be included, to establish, in consultation with the health authority, donors and the relevant professional societies, criteria for acceptability and tests of second-hand equipment to be used in radiological practices in medicine. The criteria should address the subsequent quality control and maintenance in order to ensure protection and safety while in use.
12. A requirement on registrants and licensees should be included to inform patients on risks associated to medical exposures, in a way that is commensurate to the risk specific procedures, following the risk-informed graded approach proposed in point 4.

5. Discussion and conclusions

This study has analyzed the degree of transposition and application of the International Basic Safety Standards in the FORO's countries, has identified potential gaps in infrastructure, which pose obstacles for users to comply with regulatory requirements and has provided solutions to these weaknesses. The main findings of the study are the followings:

- Division of responsibilities, ambiguities in the competences of regulatory bodies, and gaps in national infrastructure are a major challenge to ensure compliance with requirements. If the ambiguities in the law and gaps in infrastructure are too large and look overwhelming to regulators governments are responsible to look into these obstacle and assign responsibilities. On the other hand, the regulatory body needs to be proactive in searching for mechanisms to overcome these difficulties and to vigorously move towards ensuring compliance with regulatory requirements.
- Most weaknesses of the regulatory system and gaps in infrastructure can be overcome with a framework of institutional relations and cooperative agreements with other governmental organizations, such as health, education and labour authorities, calibration laboratories and custom authorities, as well as stake holders such as professional bodies, manufacturers. These relations and agreements should put in place mechanisms to enable users to comply with regulatory requirements. In particular, intense and continuous collaboration of the regulatory body with the health authority is crucial for patient protection.
- The regulatory body should use its limited staff and resources in an effective manner following a risk-informed graded approach. This approach provides a good sense for priorities in terms of risks, "devoting more effort where it is more needed", choosing the type of facilities and sources to initiate the inventory, the degree of detail for authorization, frequency and intensity of inspections, enforcement and follows up on safety deficiencies.
- The graded approach is not only important for regulators, but it is equally important for requirements on users. The need for presence of qualified experts, and patient exposure control, double-check procedures, quality control and audits, is higher in facilities with higher risk, such as therapeutic applications and interventional procedures.
- There is a delicate balance between regulating / monitoring medical exposure and interfering with medical practice. A sensible approach, based on performance oriented regulations, combined with guidance and cooperation with professional societies on protocols, quality assurance and audits, may achieve effectiveness with a minimum of interference. The common-sense approach is to do

what is really necessary. The principle of “doing more good than harm”, is also applicable to enforcement actions.

- It is not sufficient to monitor “what” is done at the facilities; it is also important “how” it is done, the degree of commitment by administrators, managers and workers, their attitudes and other indicators which bring provide confidence on safety well before any enforcement action becomes necessary, are essential items.
- Safety is not only mere compliance with requirements. It is also doing the job with full knowledge, attention, alertness, due thought, positive attitude and a proper sense of accountability. All these features together are elements of a good safety culture. The regulatory body needs to be proactive by means of a policy and a system for providing and exchanging safety information to and with stakeholders. This approach may anticipate potential problems and help remove them before they become safety threats.

With these elements in place and effectively implemented, the strength, credibility and confidence in the regulatory system may be achieved

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Radiation Protection Programmes for Transport of Radioactive Material

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Radiation Protection Programmes for Transport of Radioactive Material

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Abstract. The objective of this paper is to indicate principles for implementing radiation protection programmes (RPPs) applied to the safe transport of radioactive materials according to IAEA's Transport Regulations, Safety Standards Series N° TS-R-1. It is included a brief description of the main parts of the IAEA's safety guide TS-G-1.3 "Radiation Protection Programmes for the Transport of Radioactive Material". Consignors, carriers and consignees establish and apply systematically RPPs as a way to control radiological protection during different steps of transport activity. An example of an RPP for an organization involved in road transport of Type A packages containing radiopharmaceuticals is developed. This kind of packages is the most widely transported in the world. The scope of the RPP for the organization, considering its activities involved in transport as well as the radionuclides, quantities and activities transported, is defined. Next stage is the determination of roles and responsibilities for each activity. Taking into account the average of transports made in the last years, the type, category and quantity of packages, the radionuclides, the frequency of consignments and how long are the storages, it is evaluated an approach to the dose received by workers and special measures intended to optimize the protection. RPP is based on dose assessment and optimization, monitoring, control of surface contamination and segregation measures. The RPP also indicates main measures to follow in case of emergency during potential transport accident scenarios. Basis for training personnel involved in handling of radioactive materials to insure they have appropriate knowledge are considered. The RPP is a part of the management system of the organization and therefore it is subjected to requirements and work instructions including control, evaluation, validation, edition, revision and treatment of non-conformances. Conclusions of the paper remark the importance of implementing RPP systematically for activities related to transport of radioactive materials considering all transport players.

KEYWORDS: *transport; radioactive material; radiation protection programmes.*

1. INTRODUCTION

The objective of the present work is to indicate rules for the establishment and the use of Radiation Protection Programmes (RPPs) applied to the safe transport of radioactive material, according to the requirements of the Regulations for the safe transport of radioactive material of the International Atomic Energy Agency (IAEA).

The RPP must be established and applied in a systematic form by consignors, carriers and consignees, to consider the measures of radiation protection and their appropriate control during the stages of transport of radioactive material. In particular, in this paper it is analyzed the RPP applied to the operative stage, because it can be considered as one of the most important documents to use since it summarizes the necessary radiation protection assessments and controls for transport. It is also analyzed the importance that this document is developed on the base that converge in it the analyses, assessments and data that have been kept in mind during the previous stages of the design of packages and manufacturing of packaging, the types and quantities of packages involved in shipments, as well as of considering the quantities and frequencies of shipments, the modes of transport, etc.

2. REGULATORY FRAMEWORK

It is important to mention the regulatory framework on which it is based the necessity of establishing and applying RPPs.

Starting from the 1996 Edition, the IAEA Transport Regulations [1], and then in the 2005 Edition [2], makes reference to the International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources [3] of the Agency for the compliance with general principles on radiation protection that reflect world-wide consensus. Among other relevant requirements related to radiation protection, the mentioned Regulations require the establishment of an RPP.

In the development and application of the RPP it is useful the consideration and employment of some applicable IAEA's Safety Guides. Among those that contain specific guidelines about transport safety can be mentioned the Advisory Material for the IAEA Regulations for the Safe Transport of Radioactive Material [4], Planning and Preparing for Emergency Response to Transport Accidents Involving Radioactive Material [5] and Management System for the Safe Transport of Radioactive Material [6] and Compliance Assurance for Competent Authorities Regulating the Transport of Radioactive Material [7]. Additionally, among radiation protection related guides it is important to mention the Occupational Radiation Protection [8] and those of Assessment of Occupational Exposure due to both External Sources of Radiation and Intakes of Radionuclides [9] [10].

In particular, the Safety Guide TS-G-1.3 "Radiation Protection Programmes for the Transport of Radioactive Material" [11] is addressed to provide fundamental and detailed guidelines for an appropriate establishment and application of such programmes. This paper is mainly based in concepts included in the mentioned guide that was published by the Agency in October 2007.

3. PRINCIPLES OF RADIATION PROTECTION

It is considered important to take into account the principles of radiation protection established in the International Basic Safety Standard that are also applied to the RPPs. These principles are:

- a) Justification of the practice: no practice shall be adopted unless it produces a positive net benefit.
- b) Limitation of the dose and risk to individuals: exposures of individuals shall be less than the limits and restrictions of doses for workers and members of the public.
- c) Optimization of protection and safety: the magnitude and likelihood of exposures and the number of individuals exposed shall be kept As Low as Reasonably Achievable (ALARA principle), economic and social factors being taken into account.

4. CONCEPT AND OBJECTIVES OF THE RPP

The IAEA Regulations defines RPP as systematic arrangements, which are aimed at providing adequate consideration of radiation protection measures. Conceptually, the RPP is addressed to provide and document in a systematic and structured manner the framework of controls applied by a transport organization to satisfy the radiation protection principles and requirements; in particular, to limit exposures of workers both in normal conditions and in potential accidents. When it makes reference to a transport organization, this can mean, among other, consignors, consignees, carriers, and port operators.

The objectives of the RPPs are the following:

- a) To provide for adequate consideration of radiation protection measures.
- b) To ensure that the system of radiation protection is adequately applied.
- c) To enhance safety culture.
- d) To provide practical measures for meeting radiation protection objectives.

5. NECESSITY AND SCOPE OF THE RPPs

The IAEA Regulations require the establishment of RPPs including all the operations and conditions associated with and involved in the transport of radioactive material, including routine, normal and accident conditions of transport. The biggest emphasis should be put in operative stages of transport carried out by the transport organization and that give place to radiation exposures, such as preparation, consigning, loading, manipulation, carriage, in-transit storage, unloading, receipt at final destination and maintenance and inspection of packaging and packages.

To define the scope of the RPP it is necessary that transport organization keeps in mind:

- a) Type, nature and volume of radioactive materials to be transported;
- b) Type and category of the packages to be dispatched;
- c) Frequency to carried out shipments and other transport operations;
- d) Magnitude and probability of radiation exposures due to transport operations;
- e) Number of workers probably involved, duration of operations and distance between workers and the radioactive material.
- f) Cases of shipments carried out under exclusive use or special arrangement.

The RPP is focused totally to the most general aspects of radiation protection. However, packages containing fissile material require special considerations about criticality safety, which are outside the scope of this paper.

6. BASIC ELEMENTS OF THE RPP

The RPP should cover, with the appropriate level of detail, the following basic elements that contributing to protection and safety:

- a) Scope of the programme.
- b) Roles and responsibilities for the implementation of the programme.
- c) Dose assessment.
- d) Dose limits, constraints and optimization.
- e) Surface contamination assessment.
- f) Segregation and other protection measures (for example, packages requiring prevention of damages caused by heat or its appropriate dissipation, or packages containing material with other dangerous properties or fissile material, or Type B(M) packages requiring intermittent venting, or shipments under exclusive use or special arrangements).
- g) Emergency response arrangements.
- h) Training.
- i) Management system for the safe transport of radioactive material (also named quality assurance or quality management systems).

It is important to mention the responsibilities of the main actors involved in the RPP. The main responsibility of the transport organization is the identification and documentation of the safety and performance objectives, and the provision of organizational infrastructure and necessary resources to assure that the objectives of the RPP are properly met. The main responsibility of the competent authority is to verify independently the compliance with all applicable requirements and regulations, including the optimization of radiation exposure and safety. It is necessary that between both mentioned actors an appropriate respect and harmonized relationship are preserved, maintaining each one its essential independence.

7. REVIEW AND REVISION CYCLES OF THE RPP

Table 1 provides a possible graded approach to determine the appropriate level of detail of basic elements of the RPP. The first column includes such elements and the other three the

occupational dose as values of individual effective dose, according to the following categories:

- a) If it is most unlikely to exceed 1 mSv in a year, neither special work patterns nor detailed neither monitoring nor dose assessment programmes nor individual record keeping shall be required.
- b) If it is likely to be between 1 and 6 mSv in a year, a dose programme via work place monitoring or individual monitoring shall be conducted and appropriate records shall be kept.
- c) If it is likely to exceed 6 mSv in a year, individual monitoring shall be conducted and appropriate records shall be kept.

There are different factors that determine the importance of each basic element of an RPP, such as dose rate, radioactive contents and quantity of packages annually transported. The type and importance of the control measures to be used in an RPP should be related to the magnitude and likelihood of radiation exposures. Safety guides mentioned in “Regulatory framework” of this paper are very useful in developing those elements.

Dose and surface contamination assessment are two of the relevant items within the RPP framework. In particular, it should be considered the following: a preliminary assessment of the dose and number of workers that could be involved; the radiation monitoring of packages, vehicles, work places and workers; pertinent dose limits and restrictions; and optimization.

They are necessary successive review/revision cycles to adjust RPP to practice. It is considered that a possible strategy based on test and error can be used in the development, establishment, implementation and modification of the RPP.

Table 1. Basic Elements of the RPP and occupational dose

	OCCUPATIONAL DOSE ⁽¹⁾		
	No more than 1 mSv in a year	More than 1 mSv and no more than 6 mSv in a year	More than 6 mSv in a year
a) Scope	Yes		
b) Roles/Responsibilities	Yes		
c) Dose assessment	Radiation monitoring is not required	Radiation monitoring in the work place	Individual Radiation monitoring
d) Dose limits, restrictions, and Optimization	Yes, But basic optimization	Yes	
e) Surface contamination assessment	It shall be considered		
f) Segregation and other protection measures	Applicable to II-YELLOW, III-YELLOW, III-YELLOW under exclusive use categories; included the transport under special arrangement and packages containing fissile material or requiring special operational conditions ⁽²⁾		
g) Emergency response	It shall be considered		
h) Training	It shall be considered		
i) Quality Assurance	It shall be considered		

(1) For each element of the RPP it should be used a graded approach as appropriate.

(2) It can also be applied to packages requiring prevention of damages caused by heat, or appropriate dissipation of heat, or containing materials with other dangerous properties, or Type B(M) packages requiring intermittent venting.

By way of giving two examples of RPP, they are considered two opposed cases. On one hand, operations involving a limited number of Excepted packages, and Industrial or Type A packages with I-WHITE category, could require a reduced RPP. In that case, they are not required special operational arrangements, nor monitoring, nor segregation, but it is required basic optimization. Additionally, it is necessary to consider adequate requirements for selection and classification of packages; for marking, labelling and placarding of packages, overpacks, freight containers and vehicles; as well as for surface contamination, emergency response and management system. On the other hand, it is necessary to develop a detailed RPP and appropriately qualified human resources that can apply it for operations involving Type B(U), Type B(M) or Type C packages, or packages with II-YELLOW, III-YELLOW or III-YELLOW under exclusive use categories, including special arrangement consignments and packages containing fissile material or requiring special operational conditions. For it, the RPP requires work place or individual monitoring, detailed optimization and segregation measures, as well as specific procedures for loading, tie-down, accumulation, stowage, in-transit storage and unloading of packages. In addition, it should be considered the application of special procedures for emergency response and management system, as well as for training of transport organization personnel and drivers.

8. APPLICATION OF AN RPP FOR THE TRANSPORT OF PACKAGES CONTAINING RADIOPHARMACEUTICALS

It is briefly developed an application of an RPP for the transport of Excepted and Type A packages containing radiopharmaceuticals by road. This case represents packages worldwide used in shipments of radioactive material and the most frequently used in transport.

8.1. Scope

A company performs activities related to the transport of radioactive material and for complying with IAEA Transport Regulations, TS-R-1, has recently developed the RPP. The scope of this RPP includes the annual transport and in-transit storage of about 15000 packages containing radiopharmaceuticals, including mainly iodine compounds and molybdenum-technetium generators. About 10% of the total are Excepted packages and 90% are Type A packages. In later case, 25% are III-YELLOW category, 35% are II-YELLOW and 40% are I-WHITE approximately. The maximum Transport Index (TI) of the packages transported is 4.2 and they represent only a small fraction of packages of III-YELLOW category. The same company is performing the distribution of packages by road in its own vehicles. The air transport is not considered in this example.

It is noted that a certified supplier provides pertinent components of packaging to be used by the company for radiopharmaceuticals transport. Particularly, for components of Type A package design, the supplier has demonstrated its ability to withstand routine and normal conditions of transport for liquid and solid radioactive contents.

8.2. Roles and responsibilities

The responsible for the application of the RPP is a person trained in radiation protection that verifies the fulfilment of the requirements of the programme, including among other, training of workers, application of appropriate procedures, and worker exposure assessment and individual or work place monitoring if necessary, and emergency procedures.

The roles of personnel responsible for preparing the consignment are the verification that each package and the consignment comply with: the description of material, radionuclides and activities; type of packages; consignor's declaration; appropriate labelling and marking of packages; conformity with contamination limits; emergency cards; and loading, stowing and tie-down conditions of the vehicle. This company carries out the distribution of the following

maximum activities: 9.8 GBq of I-131; 2.5 GBq of I-125; 230 GBq of Mo-99/Tc-99m generators.

The driver's role is to know information on the applicable procedure for emergency response, and to verify conditions of loading, tie-down and placarding of vehicle, and dose rates in contact and at 2 m of the loaded vehicle.

8.3. Dose assessment and optimization

In order to identify the level of probable individual exposures and to determine monitoring requirements, the responsible of radiation protection of the company evaluates the type and quantity of packages to be manipulated, their categories and TIs, the radionuclides to be dispatched, the frequency of shipments and the duration of in-transit storage previous to dispatch the load. As a result of that preliminary assessment, the responsible of radiation protection obtained that a cargo worker could receive a maximum dose rate of 6 mSv in a year approximately, while a driver would receive 3 mSv in a year and the rest of the personnel would receive less than 1 mSv in a year. Therefore and taking into account that it is expected about 6 mSv in a year in the critical group, the responsible of radiation protection conservatively decides that a work-place and individual monitoring shall be required and appropriate records shall be kept. Then, it shall be used appropriate personal and area dosimeters, and contamination monitors (with their calibration controlled to established periods, i.e. monthly, by their manufacturers). That calibration shall be developed in recognized laboratories and certified by competent organizations. Packages, vehicles and work places monitoring will allow verifying preliminary assessment and to modify or adjust it, as applicable.

In order that doses comply with ALARA principle, the RPP requires that during operations the segregation distances can be increased as much as possible, in proximities of packages should be minimized the exposure time of workers, during storage activities can be used mechanic means of transport instead of manipulation, and packages should be maintained in the store as much as possible before to their dispatch.

8.4. Surface contamination assessment

Before the transport and also considering that radioactive materials are not in special form, the responsible of radiation protection shall verify that the contamination of external surfaces of packages and work places are below the limits. For it, they will be used sweep tests and monitors to measure the surface contamination.

The contamination of vehicles and work places shall be verified weekly, and for vehicles shall also be verified previous to their use with purposes other than transport of radioactive material. The obtained values of contamination shall be recorded. Related to equipment calibration, it shall be applied the same requirements that in precedent item 8.3.

8.5. Segregation and other protective measures

The storage area is located at 15 m of the offices, where personal of the company is considered as a member of public. Packages are stored 8 hours per day as maximum. For it, the maximum value of the sum of TIs could be limited at 10, corresponding to an annual dose of less than 1 mSv for those personnel. However, keeping in mind that the walls of the storage have 35 cm thickness of concrete, the dose would decrease in a factor of not less than 120, and therefore the personnel's annual dose will be 30 μ Sv approximately.

Of being possible, all packages and especially those with high TI shall be located at the back of the load compartment of each vehicle. It can also be used a plate of lead of about 3 mm to

provide appropriate shielding to drivers placed on the vehicle cab. Both measures will minimize driver's exposure and will comply with ALARA principle.

Packages with high TI shall remain in the storage as far as possible and they should be the last one in being loaded. It will decrease the dose on loading charge personnel complying with ALARA principle.

8.6. Emergency response

In case of an accident (high drops, severe impacts and fires) during storage or consignment loading in the vehicle, the responsible for the application of the RPP shall verify the implementation of the following measures: to give the first aids to uneven people; to maintain the public as far as possible from the area of the accident; to evaluate the fire risk and use of the fire extinction systems; to communicate with the radioprotection advisor requesting help; to maintain open communication lines; to decontaminate the pertinent area and to isolate damaged packages; to confirm that the affected area is again under conditions of being used; to restart normal operations; to make pertinent arrangements for waste disposal; to inform to the competent authority of occurred incident.

8.7. Training

The responsible for the application of the RPP shall verify that the personnel of the company involved in the preparation of the packages containing radiopharmaceuticals for transport has received the appropriate training according to the tasks that they will carry out. In that sense, the mentioned personnel shall be qualified in the following topics: transport documentation; preparation of packages; dose rates measurement and calculation of TI; labelling and marking of packages; placing and tie down of the load onto the vehicle; package segregation; placarding of vehicles; emergency procedures.

Training programs shall fulfil the applicable requirements of the competent authority regulations and the company's policy. The personnel shall be subject to biannual retraining.

8.8. Management System

The RPP is part of the Management System of the company and, therefore, it shall be subject to pertinent requirements, procedures and work instructions, including the development, control, evaluation, publication and revision, as well as non-conformances pursuit.

8.9. Review and revision cycles of the RPP

The responsible of radiation protection decides that it is necessary to implement the RPP through successive review/revision based in a test and error strategy.

The cycle starts with an input indicating the development of the RPP that is based on a preliminary assessment of their basic elements. Then, the current behaviour of the RPP should be proven to verify the compliance with the applicable requirements related to protection and safety. During the application of the RPP can be detected and recorded their achievements, deviations and non-conformances. In a next stage, it will be carried out continuously periodical review/revision and, when the deviations justify it, the revision of the RPP shall be made.

The necessary feedback should allow the RPP improvement and update, and it could imply the definition of a new scope and to develop the re-assessment. Finally, a new input generates another application of the modified RPP. This cycle should repeat periodically, the first one after a year of application and the successive cycles every three years.

9. CONCLUSIONS AND RECOMMENDATIONS

On the base of the experience acquired in the development and application of RPP, it is considered of importance to note the following conclusions and recommendations:

- Transport organizations should establish, apply and improve their RPPs in accordance to the IAEA Safety Guide TS-G-1.3 “Radiation Protection Programmes for the Transport of Radioactive Material”.
- It is considered important the publication of Safety Guide TS-G-1.3 in all IAEA official languages.
- Transport organizations should keep in mind common sense and simplicity in developing the RPP to facilitate its application.
- The review/revision cycle of RPP should be based in test and error strategy through the implementation of an appropriate Management System.
- Denials and delays of shipments do negatively impact in the implementation of the RPP.

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Management System for Regulating Transport of Radioactive Material

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Management system for regulating transport of radioactive material

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Abstract: The objective of this paper is to describe the main characteristics of the Nuclear Regulatory Authority (Autoridad Regulatoria Nuclear – ARN) management system applied to the transport of radioactive material, in Argentina. In the frame of ARN’s quality policy, “Protection against ionizing radiation on transport of radioactive materials” was selected as one of the regulatory processes, named TMR from now on. ARN’s management system is integrally based on ISO 9000 system addressed to help organizations in designing and implementing their quality management systems. TMR process was split into five sub processes in order to facilitate the implementation of the system. Such sub processes were defined taking into account of the main functions developed by ARN in the branch of safe transport of radioactive materials. For each of this processes were specified their objectives, inputs, activities and outputs, clients and stakeholders, responsibilities, supporting documents, control of documents and records, control of non-conformances, monitoring and measurements, audits, feedback and improvement. Supporting documents for sub processes were issued, validated, reviewed and improved as an essential point to achieve continuous improving. Simultaneously, some indexes were defined to monitor and measures sub processes as a way to show objective evidence of conformity with objectives. Finally, as conclusions of this paper, they will be showed the main obstacles and troubleshooting found in the design and implementation of management system as well as their solutions and state of advance.

KEYWORDS: *transport; radioactive material; management system; quality assurance.*

Background

Nuclear Regulatory Authority (ARN) is the competent authority for regulating transport of radioactive material (RAM) in Argentina. This regulation is intended to ensure protection and safety of people, property, and the environment from the effects of ionizing radiation during the transport of RAM. Such regulation is made effective through the application of National Standard AR 10.16.1 “Transport of radioactive materials” [1] which has been literally taken from the 1996 Edition (Revised) of IAEA Regulations TS-R-1 “Safe transport of radioactive material” [2]. Both of the standards, AR 10.16.1 and TR-S-1, will be referred hereafter as the Transport Regulations provided that there are no differences between them.

In 2002 ARN signed an agreement with National Technological University with the purpose of cooperation and advice in establishing a Quality Management System. The objectives were to get higher efficiency and effectiveness, continual improvement of regulatory and supporting processes, as well as to ensure the best information for citizen and the transparency of ARN actions.

Based on these previous conditions and sure of having developed its regulatory experience with a high level of confidence, the ARN decided to implement its Management System (MS) in 2005. Through a directory’s resolution N° 56/05 they were defined eleven processes, seven of them are regulatory process and the other four are supporting processes which are aimed at support the regulatory objective and tasks of ARN. The term management system reflects and includes the initial concept of “quality control” and its evolution through “quality assurance” and “quality management”

ARN’s management system

The ARN’s objectives and working system for implementing, monitoring, and improving work are established in the Quality Manual. It is based on standard ISO 9001:2000 “Quality management systems – Requirements” [3] and follows the recommendations of “PDRP-6 – Quality management of the nuclear regulatory body” from IAEA [4].

The MS covers all activities and facets of the organization, which is split in regulatory and supporting processes. The Quality Manual establishes that ARN fulfills its mission performing, among others, the following actions:

- identifying the necessary processes of the MS and its applications through the organization,
- determining the sequence and interaction of these processes,
- determining criteria and methods to ensure efficiency and effectiveness in the management of each process,
- ensuring availability of resources and information to support the processes,
- performing the monitoring, measurement and evaluation of the processes, and
- implementing the necessary actions to achieve planned results and continuous improvement of the processes.

The present paper focuses the regulatory process named “Protection against ionizing radiations in transport of radioactive material”, indicated as TMR Process hereafter. The objective is to assure protection to people, properties and the environment performing an adequate control on the effects of ionizing radiations, controlling the compliance with the Transport Regulations and applicable requirements to transport of radioactive materials, performing safety evaluations, issuing necessary requirements as result of inspections and regulatory audits, implementing preventive and corrective actions to control radiological hazards, criticality and effects of heat.

TMR Process

This process applies to the transport of radioactive material by all modes on land, water or air, including transport which is incidental to the use of the radioactive material. Transport comprises all operations and conditions associated with and involved in the movement of radioactive material; these include design, manufacture, maintenance and repair of packaging, and preparation, consigning, loading, carriage including in-transit storage, unloading and receipt at the final destination of loads of radioactive material and packages.

TMR process was split into five sub processes. Stakeholders were defined for each one of these sub processes including National State, society, organizations, other ARN’s sectors and users. In the context of this paper, user means a person or organization, who designs, tests, assesses, manufactures, services, maintains, consigns, carries or otherwise uses a package in connection with the transport of RAM.

The mentioned sub processes are defined below:

1. Development and updating of regulations and regulatory guides.

It is included the edition and updating of national standards, regulatory guides and other type of documents such as forms and instructive docs to assist customs involved in transport of RAM. In the same way, it is considered the participation in drafting, modification and update of international standards and guides related to transport of RAM.

2. Licensing of packages, special radioactive materials and radioactive material shipments.

Since the safety in transport of RAM has a strong dependence on the design of packages and materials, and on carry out of certain shipments, the licensing of such designs can be a complex process that requires many different skills. The Transport Regulations provide the elemental rules for design and it

requires expert interpretation because the implications of regulatory interpretation can have important consequences on the design process.

3. Compliance assurance during transport of radioactive materials.

To ensure compliance with Transport Regulations it is essential to demonstrate that safety is properly taken into account. This is performed through a program of inspections and audits. For each type of inspection, a number of procedures were written, which include items to be taken into account, proper way to perform the inspection, measurements, evaluations, documents to be required and directives for the final report.

4. Advice, training, and communications.

Careful considerations are given to assist stakeholders in the interpretation and application of the Transport Regulations. This sub process also includes development and sponsoring of seminars, conferences, workshops, training courses for personnel involved in the subject. Information in the form of notices and guides that are intended to spread news and the state of art related to radiological safety in transport of RAM is periodically distributed.

5. Support and advice in emergencies involving RAM

Specialized teams having the support of the entire ARN organization perform intervention in radiological and nuclear emergencies. Inside such structure this sub process needs to be prepared and available to act in the case it is required.

Quality plan

A quality plan for TMR Process was developed in the context of the established management system of ARN. The indications of ISO 10005 [5] were followed to write the plan, which was intended to organize and manage activities to meet quality requirements, to optimize the use of limited resources of the organization and to be used as a basis for monitoring and assessing compliance with the requirements, both internal and external. Tasks as control of documents, records and resources including materials and human resources as well as infrastructure and work environment, customer communication, purchasing, control of non conformances and audits are accomplished under the umbrella of ARN's MS. Own documented procedures were written to assure the requirements are met. Such written procedures cover production and service provision, identification and traceability, customer's property, preservation of product, monitoring and measurement.

Documented procedures

In order to establish a systematic way to perform the work, procedures were written about the following subjects:

- Issuing of approval certificates
- Analysis and assessment of applications
- Inspections to consignments and shipments of RAM
- Verification of requirements before the first shipment
- Inspections to tests for demonstrating ability to withstand normal and accidental conditions of transport

- Inspections to tests for special form radioactive material
- Performing of schedule for inspections to consignments and shipments of RAM
- Advice and communication
- Training

Performance indicators

In general, performance indicators help illustrate how well the organization is doing in meeting its objectives or achieving the desired outcomes. Particularly, performance indicators were developed as a way to measure the accomplishment of TMR Process' goals such as performed vs. scheduled inspections, issued vs. required approval certificates, time of issuance of approval certificates, update of databases, update of standards and regulatory guides, advice and satisfaction of stakeholders.

The figure 1 illustrates the map of the TMR process while main basic elements of each sub process are shown in Table 1.

ISO 9001:2000 certification

IRAM and IQNet certificates were obtained on 20th May 2008, after the certification audit addressed to TMR Process. They have three years of validity with a previous maintenance audit one year after the certification.

Conclusions

The MS plays an important role in the efforts of the Competent Authority to obtain as far as possible full assurance of compliance with the Transport Regulations as well as to assist in self-correction and self-improvement. In addition, the application and use of a MS can promote public confidence in the transport of RAM.

It is important to remark main difficulties and achievements found in the application of the MS to regulate transport of radioactive materials. On the side of problems, there were some difficulties in implementing general procedures when responsibilities from different sectors were involved and the interfaces among those sectors and their tasks had to be reformulated. In addition, in some cases it was difficult to find representative indicators to measure the sub processes. On the other side, it is necessary to recognize that several important achievements were reached such as replica of knowledge and experience, greater vision of the system, better information management, better organization and perception of continuous improvement.

The long-term success of MS will depend on whether all individuals have an appropriate awareness and understanding of the objectives, principles and benefits of the MS functioning in the organization.

Figure 1 - TMR Process map

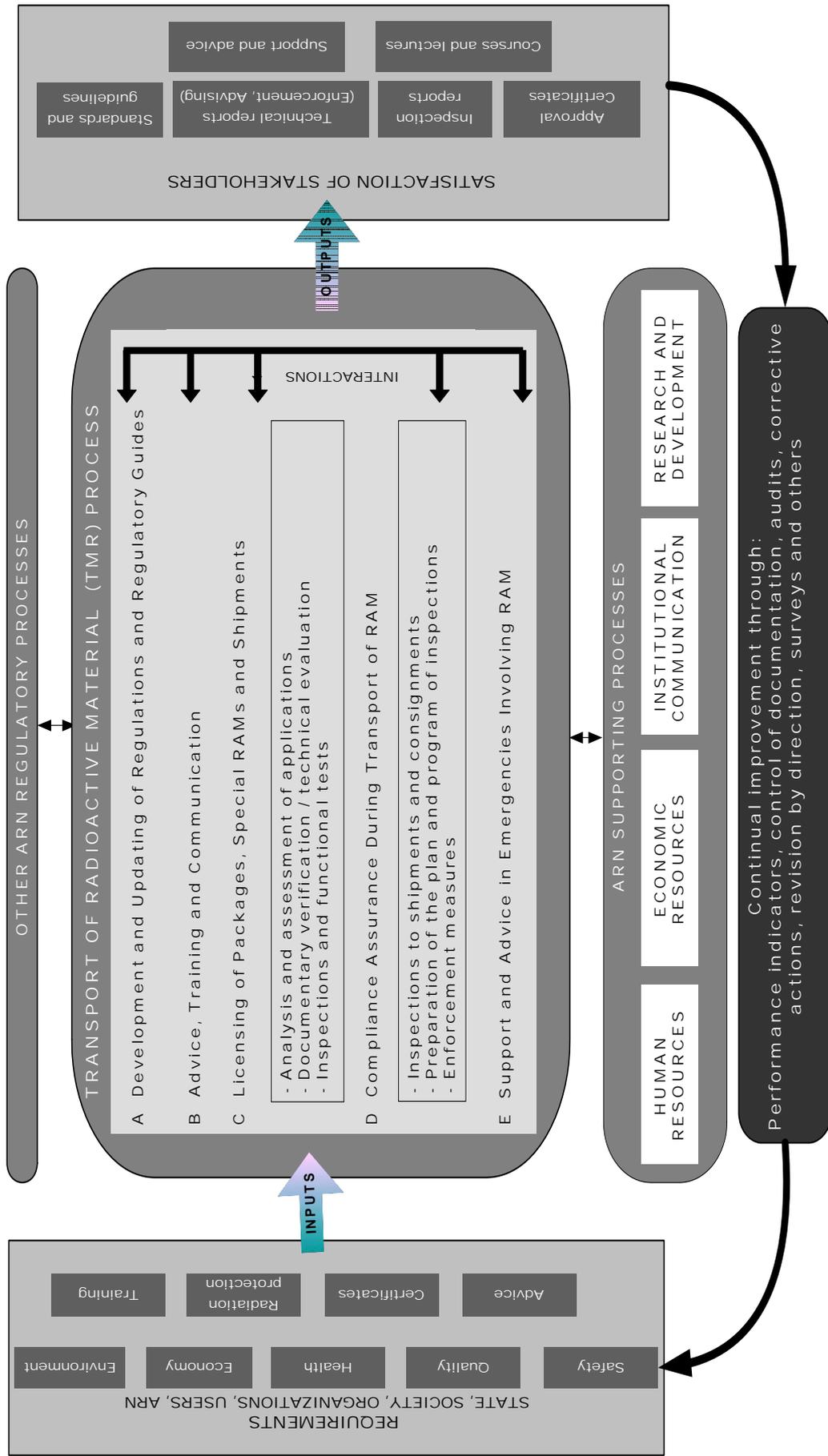


Table 1: Basic elements of sub process of the TMR Process

Sub process	Input	Control point	Non conformances	Performance indicator	Objective	Output
Development and update of standards and regulatory guides	<i>National law, conventions, threats, international recommendations.</i>	<i>Revision of standards and guides.</i>	<i>Mistakes in documents.</i>	<i>Update of standards and guides.</i>	<i>< 1 year.</i>	<i>Standards and guides.</i>
Licensing of packages, special radioactive materials and radioactive material shipments	<i>Application for certificates.</i>	<i>Legal, administrative and technical coherence.</i>	<i>Mistakes in certificates. Delays in issuing of certificates.</i>	<i>Quantity of emitted certificates vs. applications. Time of issuance.</i>	<i>100%. < specified time according to the type of certificate required.</i>	<i>Approval certificates.</i>
Compliance assurance during transport of radioactive materials	<i>Program of inspections, notifications, applications import/export.</i>	<i>Compliance with program, equipment calibration, application check.</i>	<i>Procedures not followed.</i>	<i>Inspections performed vs. programmed. Update of databases.</i>	<i>100%. 100%.</i>	<i>Inspections, reports, verified forms of import/export.</i>
Advice, training, and communications	<i>Requirement for advice, training or information.</i>	<i>Surveys, examinations, peer reviews.</i>	<i>Advice, training or information not given.</i>	<i>Satisfaction.</i>	<i>80%</i>	<i>Advice, training, information.</i>
Support and advice in emergencies involving RAM	<i>Notification SIEN / SIER.</i>	<i>ARN's procedures.</i>	<i>Procedures not followed.</i>	<i>N/A</i>	<i>Defined by SIER / SIEN.</i>	<i>Assistance to SIER / SIEN.</i>

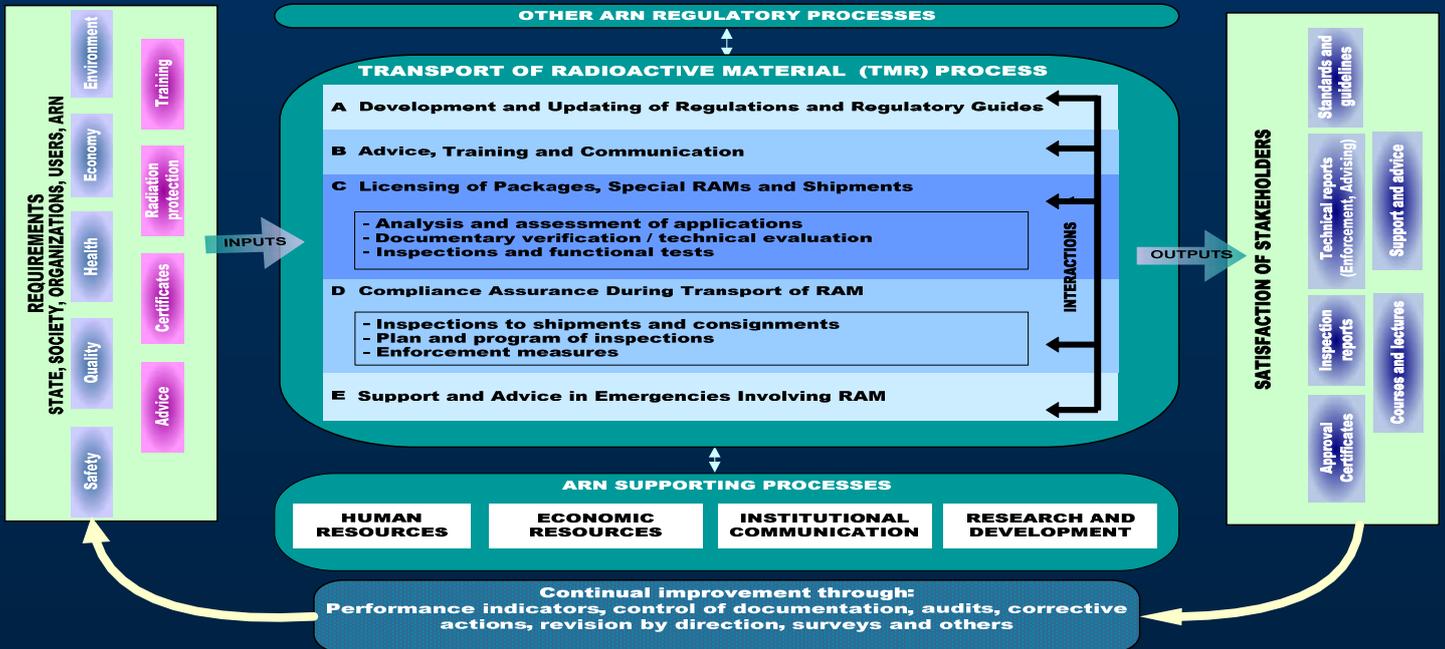
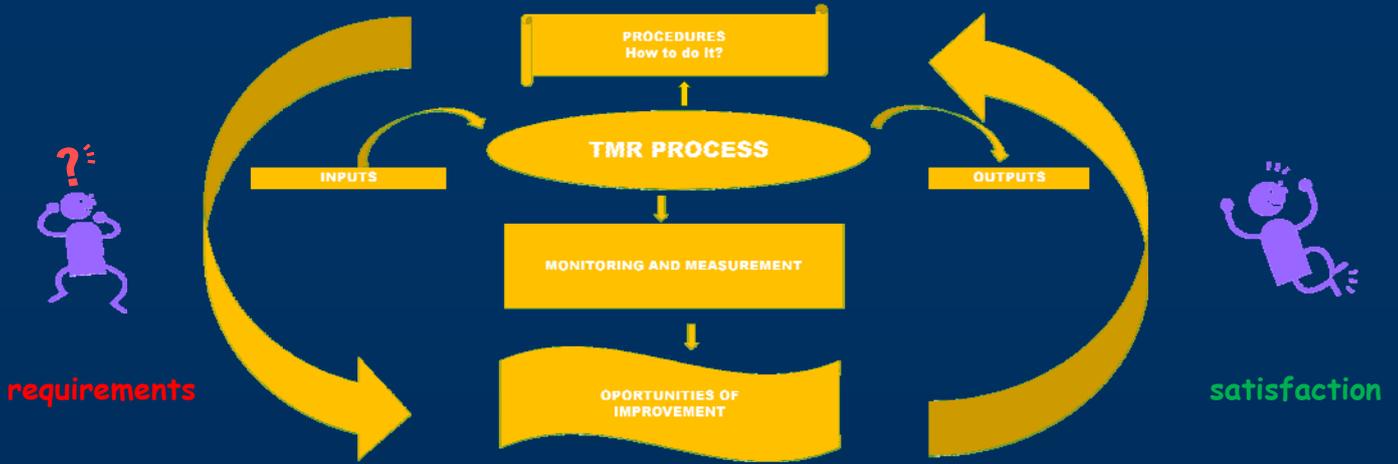
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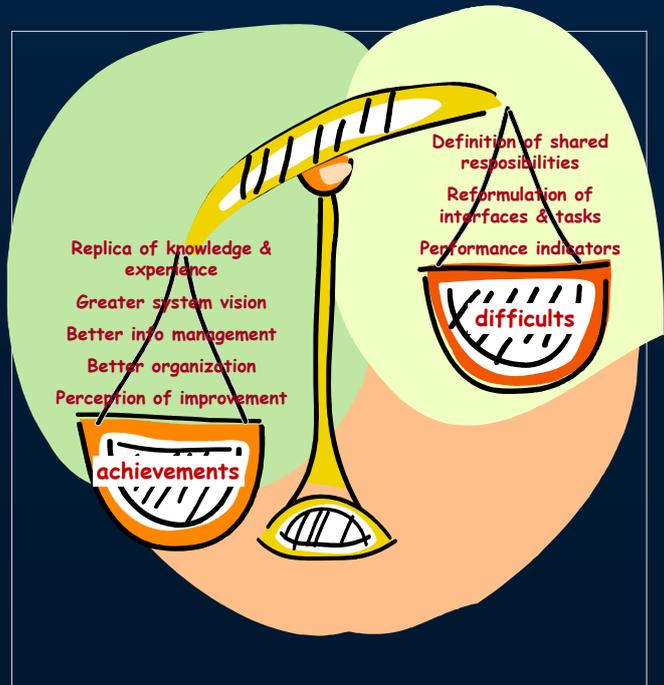
MANAGEMENT SYSTEM FOR REGULATING TRANSPORT OF RADIOACTIVE MATERIAL

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ISO 9001:2000 CERTIFICATES



Transport of Cobalt 60 from the Argentine Nuclear Power Plant in Embalse

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Este trabajo se presentó oralmente durante el evento.

TRANSPORT OF COBALT 60 FROM THE ARGENTINE NUCLEAR POWER PLANT IN EMBALSE*

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Abstract

The purpose of this presentation is to point out some relevant issues related to the transport of cobalt 60 for domestic uses and export from the Argentine Nuclear Power Plant located in Embalse (ENPP). It is remarked that in last 22 years, ENPP produced about 2600 PBq (70 MCi) of Co-60, and that Argentina is the third Co-60 producer, after Canada and Russia. It is mentioned that in Argentina there are facilities which manufacture Zircaloy seamless tubes and reactivity control bars containing cobalt 59 to be used in ENPP, the operation of Candu reactor at ENPP including cobalt-related activities, and the manufacture and commercialization of Co-60 sealed sources. The Argentine Nuclear Regulatory Authority verifies the compliance of nuclear activities with regulatory standards, including those related to the transport of radioactive material. Then, it is described the process for obtaining Co-60 at ENPP with an activity concentration to allow its use in medicine and industry. After that, is considered in detail the type of package used for the transport of Co-60 from ENPP: it is to say, Type B(U) package designs that must be approved by the Competent Authority origin of the design, and they are able to withstand routine, normal and accident conditions of transport, including severe impacts, fires and immersion in water. The overall dimensions of packages are about 1.0 m to 1.5 m diameter by 1.2 m to 1.8 m high, as well as the gross mass could vary from 6500 kg to 9500 kg. Radioactive contents consist of Co-60 sealed sources (called pencils), with an activity range from 8 PBq to 16 PBq. Then, it is considered activities prior to dispatch of Type B(U) packages at ENPP, including the preparation of radioactive contents, packaging and package (labelling and marking) as well as necessary tests and documentation for transport in order to fulfil pertinent requirements of package design operation manual of the package and related standards. Finally, it is pointed out the Type B(U) package tie-down system to the vehicle, and the proper placards of the loaded vehicle ready to dispatch.

* Presentación oral

Implementation of a Strengthened International Safeguards System. ABACC 15 Years

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IMPLEMENTATION OF A STRENGTHENED INTERNATIONAL SAFEGUARDS SYSTEM ABACC 15 YEARS

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- Introduction

The purpose of the paper is to explain how the system of a regional safeguard has been operating and developing in the framework of the Brazilian-Argentine Agency for Accounting and Control of Nuclear Materials (ABACC), and how the international recommendations of radiological protection must to be taken into account in the safeguards implementation and its impact in the international context.

The ABACC has been a dynamic system, which contributes worldwide in the application of the regional and international safeguards. In 2006, the ABACC celebrated its 15th anniversary. The ABACC was created in 1991 in the framework of a Bilateral Agreement for the Exclusively Peaceful use of Nuclear Energy, the ABACC was created in order to apply the aforementioned system called “Common System for Accounting and Control of Nuclear Materials” (SCCC). During this time, the ABACC has grown in its implementation and has become a model in the application of regional safeguards that is recognized internationally.

The ABACC was the pillar to signed an Agreement between Argentina, Brazil, the ABACC and the International Atomic Energy Agency, called “Quadripartite Agreement”, committed themselves to accept the application of safeguards to all nuclear materials in all the nuclear activities performed in both countries.

The ABACC and the relevant implementing and supplementary agreements, set forth the conditions for the peaceful use of nuclear energy, the exchange of technical staff, the transfer of knowledge and international cooperation in a strong commitment to non-proliferation of nuclear weapons.

1- Implementation of Local Regulations in line with the International Non Proliferation Regime (SCCC).

On the basis of recognizing the sovereign right of every nation to have access to nuclear technology for scientific, technological, economic and social development of their inhabitants, at the end of 1980s, Argentina and Brazil reaffirmed their decision to provide mutual transparency to their nuclear programs. Both countries assume the commitment to use all nuclear materials and nuclear facilities exclusively for peaceful purposes.

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It was within this context that they created the Common System for Accounting and Control of Nuclear Materials (SCCC) as a comprehensive bilateral safeguards system and the Brazilian Argentine Agency of Accounting and Control of Nuclear Materials (ABACC) as an independent regional body, in order to manage and apply it.

At the end of 1990s, the states parties and ABACC signed with the IAEA a comprehensive safeguards agreement based on the SCCC and on the INFCIRC/153 model. The application of a total safeguard system began when the Quadripartite Agreement entered in force in March of 1994.

The Quadripartite Agreement is the legal instrument that regulates the relationship between the States Parties, the ABACC and the IAEA. The essential part of the agreement is the Common System for Accounting and Control of Nuclear Materials (SCCC), in which on emphasis is made on the fact that ABACC and the IAEA are partner institutions that must work in a way to apply effective and efficient safeguards to nuclear materials.

2- Cooperation between the Regional System and the IAEA

Several points in the Quadripartite Agreement establish cooperation activities that must be performed with regard to nuclear safeguards. Following these directives, the ABACC and the IAEA have been working together in the development and improvement of safeguard approaches, in modern and secure communication systems, in inspector training and on the resolution of all the implementation problems detected.

The Argentine and the Brazilian authorities asked both agencies to coordinate their task, with the goal of an efficient management of the cost-benefit of safeguard activities. The cooperation within the framework of the Quadripartite Agreement is an on going activity and the success of a regional and an international safeguard system working together, relies on the good coordination between the parties.

3- Coordination between the National and the International Systems

International inspections are carried out in cooperation with the state counterpart, following certain fundamental premises as follow:

- ◆ The inspectors should know the radiation protection, safety, and physical protection procedures that have been implemented in the facilities being inspected.
- ◆ The inspectors should avoid the undue interference with the nuclear activities in the state.
- ◆ ABACC and IAEA inspectors should avoid the duplication of inspection activities in the field.
- ◆ The agencies should maintain confidentiality regarding the technological development information.

On the other hand, the state party assumes certain commitments that, in the particular case of Argentina, can be summarized as follow:

- ◆ The state should provide the administrative and technical support, in order to assure that either IAEA or ABACC inspectors should meet their inspection goals.
- ◆ The State party must provide access conditions that assures doses as low as possible -below the dose limits- and proper support under any potential dangerous or accidental situations.
- ◆ Both agencies should be informed in advance about any change in the current regulations or access procedures.

4- Regulation on Radiological Protection versus Safeguards Implementation

The national regulation in line with the international recommendations on radiological protection must be taken into account in the safeguard implementation. At the state level the responsibility is to preserve the fulfillment of international commitments, to update the nuclear material inventories as required and to evaluate the radiological risks and their impact on the safeguards activities and vice versa.

In the following examples we try to show the role played by the National Regulatory Body and the interaction, between the radiation protection national regulations and the safeguard implementation.

- Impact of Safeguard Activities on Radiation Protection Procedures

Following international regulations, each organization is the responsible to apply radiation protection controls to their inspectors. In this regards, each organization provides its own dosimeters and make arrangements for periodic whole body counting measurements to the inspectors, to record the corresponding doses and to detect any potential internal contamination.

In the Argentinean case, on despite of the effectiveness of the radiation protection controls implemented by each organization, an additional national surveillance is applied. Consequently, external radiation and internal contamination control by providing whole body counting, TLD personal dosimeters and excretes assays respectively can be requested, as applicable.

This procedure has been implemented in order to enable the state fulfill the responsibility assumed within the safeguard agreement and to detect, as soon as it is possible, any unexpected exposure or contamination problems that could arise in the Argentinean facilities during safeguards inspections. Even though these criteria reinforced the radiation protection controls, due to the previous coordination required, they represent difficulties on the coordination of the international inspections considering the new modalities that are being implemented.

Normally, for routine inspections the agencies provide the state party at least one week in advance notice concerning the arrival of the inspectors at the facilities and the activities to be carried out. However, a portion of these routine inspections can be performed on short notice or on unannounced basis with no more than 24 hours of advance notification.

Nowadays the short notice modality is becoming more frequently; consequently the regulatory body of Argentina has modified the radiation protection controls in order to avoid any undue interference with the ABACC and the IAEA activities.

In the framework of the responsibility assumed under the Quadripartite Agreement, the application of national regulations on individual monitoring has been reviewed in order to optimize the protection of international safeguard inspectors on duty. In this regard, excretes analysis is required on case by case basis, if applicable; whole body counting have been restricted just to those facilities handling irradiated material and TLD personal dosimeters are provided, as usual, at the beginning of the inspection program. The ARN has the right to require the international inspectors a non-routine measurement in case the results of the air and surfaces monitoring programs show abnormal results.

The immediate consequence of these modifications has been the successful implementation of the short notice random inspection regime in Argentina, during 2008, and an improvement of the cost effectiveness of IAEA and ABACC inspection effort.

- Impact of Radiation Protection Regulation on Safeguards Activities.

From the safeguards point of view, the remaining low enriched uranium content in the spent MTR fuel elements, can not be used directly for the production of nuclear weapons due to the low content on the isotope U235 and the irradiated status. Nevertheless, the safeguards criteria requires these spent fuels have to be periodically counted, identified and the irradiated status confirmed, in order to maintain the knowledge on the inventories of these type of nuclear material.

While these fuel elements are stored in the reactor pools, the access for verification purpose is normal and does not require any special arrangement. The situation drastically changes when items are sent to long term storage facilities with difficult access. Identification is not possible any more and the access to counting is highly dependent on the storage conditions.

The nuclear material is normally verified before being transferred to the long-term storage and the continuity of knowledge is maintained applying containment and surveillance measures. In spite of the fact that the results of the containment and surveillance measures were satisfactory, the inventory of nuclear material has to be periodically confirmed. Consequently, provisions should be adopted by the design to verify these items or the introduction of operational restrictions would be required to meet safeguards goals.

In the case of an old wet storage facility in Argentina, the fuel elements are introduced into underground wells filled with water. Up to two fuels elements can be introduced inside each well to meet the maximum capacity. The old design did not foresee the current verification requirements.

From safeguards point of view, the NDA measurements from the top of the well is adequate to confirm the irradiated status but do not allow counting of fuels inside the well.

Under the current conditions, ABACC has developed a methodology where the counting and the confirmation of irradiated status can be done through NDA scanning of the total length of the well. Unfortunately to do this activity is necessary the opening of the well from the top and the introduction of detectors inside it.

According to the data recorded during the field test, the dose rate measured on the top of the well filled with two fuel elements was higher than 5 mSv/h. The dose rate fell to less than 0.1 mSv/h when only one fuel was stored in the well. In addition, the risk to damage the surface of the fuels is high due to the small diameter and the limited space available.

Taking into account these facts, ARN has considered that remote handling is recommended to introduce detectors inside filled wells due to the high doses involved when maximum capacity is used. Consequently, it has taken actions on the operator and on the agencies to find an acceptable solution for all the parties involved.

On one side, ARN has requested the operator to store just one fuel per well while remaining capacity is available in order to optimize the doses. These conditions, complemented with surveillance and containment measurements do not require counting of fuels. The irradiated status can be confirmed through NDA measurement from the top of the well with detectors remotely handled. This practice is compatible with the radiation protection regulations and both agencies are able to meet their inspection goals. The problem is the operational restriction imposed.

On the other hand, the regulatory body has requested the ABACC and the IAEA to consider an improvement of the current technology as an alternative for the future, in order to remove the operational restriction in case it is needed.

Finally, ARN has suggested the operator to speed up the conclusion of the new storage with an appropriate design to support all the safeguard activities without any impact on the radiation protection regulation.

- Contribution of Safeguard Data to Nuclear Safety Controls and Environment Protection.

The common system of accounting for and control of nuclear material is a set of accounting procedures that requires the state members the development of accounting database systems. In Argentina, the regulatory body has implemented a central database and has assessed the operators on the implementation of complementary facility databases and record systems in order to fulfill the requirements of the regional safeguards system.

The accounting system implemented at facility level permits monthly updating of the inventories; provides traceable records on the waste materials and gives information on unmeasured losses or accumulated inventories in the case of nuclear material handling. In addition, the quality of the operator measurement system and the true physical values of these quantities can be confirmed at least once a year.

On spite of databases are restricted to nuclear material inventories and beside the use of this information for safeguard purposes, the reliability of these databases represent an important source of data for other regulatory activities like the licensing process and the radiation and physical protection controls. Effectively, through these data, the regulatory body is able to monitor the mass limit restrictions imposed by license, the dates of material transfers, the true inventories of nuclear material at the facilities and their waste discharges.

5- Conclusions

Even though the non proliferation regime contributes to build confidence and transparency between nations and consequently to protect the public and the environment from nuclear disasters, the verification activities have to be implemented under the framework of the universal principle of public, environment and workers protection against radiation effects.

The national regulatory body has the responsibility to require access to nuclear material under adequate radiological conditions. The current legal framework provides the legal tools to require corrective actions from the Operating Organization. On the other hand, the cooperation between the states, the regional system and the IAEA, allows the optimization of the technical capacity and economic and human resources in order to resolve the implementation difficulties and to provide credible assurance on non- diversion of nuclear materials.

The successful application of the SCCC, the atmosphere of mutual confidence and cooperation between the state members, ABACC and the IAEA, show an effective contribution to the nuclear non-proliferation regime in South America. In addition, safeguard data, verification activities and containment - surveillance measures applied with safeguard purposes, introduce additional restrictions to the access, and to any unauthorized use or movement of the nuclear materials. These actions strengthen the protection of the environment and public with an effective control on the inventories of fissile material and irradiated fuel elements.

The on going debate worldwide on the definition of energy policy centered on competitiveness, security of supply and environmental concerns provides an opportunity to consider regional coordinated actions. The Brazilian-Argentine initiative in the nuclear non-proliferation field can be taken as an example for the new challenges, in particular those related with the protection of the health and the environment.

Servicios del Centro de Información de la Autoridad Regulatoria Nuclear

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Argentina

Abstract

El objetivo de este trabajo es difundir los servicios y las actividades de la biblioteca de la ARN a usuarios del área de dosimetría biológica en el marco de las Jornadas de Intercomparación de la Red Latinoamericana de Dosimetría Biológica, realizado en Buenos Aires, del 27 al 30 de octubre del 2008. En el mismo, se hace una breve reseña histórica de la biblioteca de la ARN, se detallan los servicios que presta actualmente a cada tipo de usuario y se describen las tareas que se llevan adelante en concepto de colaboración con otras instituciones como el comité de terminología del IRAM; el ingreso de material a la base de datos INIS del OIEA; la digitalización retrospectiva de publicaciones de nuestra institución, su indización y descripción bibliográfica para el Foro de Organismos Reguladores Iberoamericanos y la participación en redes y consorcios. Finalmente se muestran algunos datos de las estadísticas anuales de la biblioteca mostrando el comportamiento de los usuarios internos de la ARN.

* Presentación oral

About the Application of “D” Values to Radioactive Waste

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About the application of “D” Values to radioactive waste

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Abstract: A system for categorizing radioactive sources and quantities of radioactive materials aimed at specifying generic security levels has been developed during the last years. The minimum activity of a given radionuclide that it is considered “dangerous” (i.e. the “D value” of such radionuclide) is derived from a set of well-defined exposure scenarios, but the low activity concentration of radioactive waste, particularly of very low level waste (VLLW), low level waste (LLW) and intermediate level waste (ILW), makes inappropriate the direct application of the D Values. Taking into account the qualitative definition of a “dangerous source” and the role of the activity concentration, a quantitative approach for the determination of the level of security applicable to the case of radioactive waste is proposed.

KEYWORDS: *D values; radioactive waste; dangerous sources; security level.*

1. Introduction

During the last years a system for categorizing radioactive sources and quantities of radioactive materials has been developed, aimed at specifying security levels. The qualitative definition of a “Dangerous Source” as well as threshold values for several radionuclides (“D Values”) were first published in 2003 [1]. Later on, these values were quoted in reference [2], incorporated in reference [3] and, finally, a publication addressing the systematic derivation of the D values was published in 2006 [4].

As it is explained below, though the qualitative definition of a “Dangerous Source” is applicable to all situations, including radioactive waste (RW), the scenarios used for deriving the “D values” cannot be directly applied to all kinds of radioactive material and, in particular, they seem not to be applicable to very low level waste (VLLW), low level waste (LLW) and intermediate level waste (ILW).

2. The Qualitative Definition of a Dangerous Source and its Implications

In the aftermath of the 11 S, concerns arose all over the world about the possible malevolent use of radioactive materials and, therefore, the need to increase the level of security when dealing with high activity radioactive sources or with significant amounts of radioactive materials. Because of the large range of activities involved and the great number of radionuclides used worldwide, not only in the nuclear industry but also in medicine, industry and research, it was necessary to define a conceptual framework explaining when a radioactive source should be considered “dangerous”. Such conceptual framework, namely the qualitative definition of a “dangerous source”, had to be formulated in such a way that it would be possible to derive an “activity limit” per radionuclide. Below this limit or threshold, the so-called “D Value”, a source or amount of radioactive material would not be considered as “dangerous”¹. Once obtained this “dangerous threshold”, security levels may be specified based on the ratio between the activity of concern and the respective D Value.

A “dangerous source” was defined as a source that could, if not under control, give rise to exposure sufficient to cause “severe” deterministic effects² and, in this context, a “severe deterministic effect” was defined as an effect that is fatal or life threatening or results in a permanent injury that decreases

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¹ And should be subjected only to prudent management practices.

² A *Deterministic effect* is expressed as tissue reactions that can be diagnosed on the individuals exposed. It is characterized by a threshold dose below which the effects are not noticeable and by the increase of the severity of the tissue reaction as the dose increases above the threshold.

the quality of life. The concept of “dangerous source” has been converted into operational parameters by defining exposure scenarios from which, for each radionuclide, it was calculated the activity that could lead to such severe deterministic effects [4].

It should be noted that the definition explicitly points out that it may be possible, in the event of misuse, the occurrence of deterministic effects. Furthermore, such deterministic effects should be high enough as to make it likely either the death of a person or a severe permanent injury. In that sense, it was considered that existing categorizations, for instance the one used for establishing the level of safety for the transport of radioactive materials [5] were not appropriate and experts considered that it was necessary to develop specific scenarios for deriving D values for each radionuclide.

Wisely, the qualitative definition of a Dangerous Source does not require other than conventional security measures when the radiological consequences to health are not significant. Such approach is intended to allow a rational use of security resources by disregarding possible malevolent acts that may have a great impact in the media without generating an actual radiological risk and, in this framework, the extrapolation of the D Values to radioactive waste – excluding the case of disused sealed sources- seems to be inappropriate.

3. The Scenarios used for deriving the D Values

As usual, exposure scenarios were developed for deriving the D Values [4]. These scenarios consider external exposure of the non-dispersed material or exposures that may result from the dispersion of the material. For each scenario several pathways were defined as well as the dose limits that may result in severe deterministic effects. The lower value of activity obtained was the threshold activity, and two activity limits were derived, one for non-dispersible material (D_1) and other for dispersible material (D_2).³

For calculating the D_1 -Values (non-dispersible material) the following scenarios were considered:

- The “pocket” scenario, in which it was assumed a person carrying an unshielded source in the pocket.
- The “room” scenario, in which it was assumed that a person remains in the vicinity of an unshielded source from days to weeks.

Regarding the D_2 -Values (dispersible material) the following scenarios were considered:

- The “inhalation” scenario, in which it was assumed that a fire or explosion aerolized a fraction of the radioactive inventory.
- The “ingestion” scenario, in which the most limiting of the following two pathways was considered: the handling of a leaking source or the contamination of a public water supply.
- The “contamination” scenario, in which it was assumed that the skin became contaminated due to a leaking source.
- The “immersion” scenario, specific for noble gases, in which it was assumed that the activity is released into a room.

4. Limitations of the Scenarios Used for Deriving the D Values when dealing with Radioactive Waste

The D Values were derived for radioactive material with high activity concentration. The fact that radioactive waste, in particular VLLW, LLW and ILW; has low activity concentration has implications both in the case where external exposure dominates as well as in the case where internal irradiation dominates.

³ For practical reasons, was recommended to consider as the D value of a given radionuclide the lowest between D_1 and D_2 .

In the case of external exposure, factors such as dimensions of the source may not be physically possible for some scenarios, for instance the “Pocket Scenario”. Regarding the “Room Scenario” it seems logic to consider only one waste package and not to add the activity of several waste packages as well as to exclude radioactive waste in bulk. In addition, the dose rate at a given distance of an unshielded waste package cannot be directly related to their activity contents because of its size and, in most cases, the significant self-attenuation.

Regarding intakes, for assessing the incorporation of radioactive material the activity concentration had to be taken into account and consideration should be given to a likely intake mass, which its is not related with the activity contents and can not be formulated as a fraction of the radioactive inventory. This is the case for radioactive waste, where the activity is distributed throughout inert material, either on the surface (e.g. contaminated paper, rubber or piping) or included into an inert matrix (e.g. concrete.)

It is stressed that the concentration of the activity had to play a key role in the derivation of security levels for radioactive waste because it makes impossible, in most cases, the occurrence of deterministic effects. The concentration of activity has been used in this sense in other cases, for instance for transporting radioactive material, where specific provision allows the use of industrial packages in the case of LLW and ILW [6], or for exempting up to about one tone of radioactive material [7]. The experts that developed the D values did not consider this element because at that time they were not concerned with the case of radioactive waste and, therefore, they adopted activity concentrations compatible with the cases under study.⁴

5. Scenarios Applicable to the Case of Radioactive Waste

A brief description of the scenarios used for deriving the D values is presented below, those applicable to the case of radioactive waste are indicated and specific proposals for its use are made.

5.1 Non-dispersible material scenarios

- **Pocket Scenario**

This scenario is not applicable to the case of radioactive waste due to the dimension of the source (e.g. 200 liters drum)

- **Room Scenario**

This scenario may be used for radioactive waste, but each waste package should be assessed individually and, for the reasons explained above, instead of using the D_1 value it is proposed to utilize the dose rate at given distance of the unshielded waste package. The same distance, exposure time and dose threshold of this scenario, though conservative, may be used for deriving the limiting dose rate. Taking the same reference level (i.e. total body dose for the onset of severe deterministic effects 1 Gy) and assuming that a person will remain 100 hours at a distance of 1m from a RW package, the dose rate threshold for considering it as a dangerous source would be 10 mGy/h.

5.2 Dispersible material scenarios

Regarding the scenarios considered when deriving the D_2 values, only the inhalation pathway seems to be applicable to the case of RW, being this scenario usually the most conservative one. As indicated above, for most RW, and certainly for VLLW, LLW and ILW, the maximum intake would be governed by the mass that may be incorporated in the case of release. For that reason, it is proposed to consider an activity concentration limit in the case of RW. Taking into account that it is most unlikely that a person would remain in a dusty atmosphere long enough as to intake more than 10 mg⁵ of

⁴ For some elements, for instance natural uranium, it was taken into account their specific activity, but in no case it was considered the radionuclide dispersed into a non-radioactive material.

⁵ It is noted that a maximum intake of 100 g is indicated as possible in reference [4]. This is a mistake, a

material [5-8], the threshold activity concentration (C_L) for considering a RW as a dangerous source can be calculated as a function of the D_2 value as follows:

$$10^{-4} \times D_2 [Bq] = 0,01 \text{ g} \times C_L \left[\frac{Bq}{g} \right]$$

where,

$10^{-4} D_2$ is the activity intake that produces the reference dose,

10 mg is the maximum intake mass of material, and

C_L is the threshold activity concentration for considering a RW as a dangerous source

So,

$$C_L \left[\frac{Bq}{g} \right] = \frac{10^{-2} D_2 [Bq]}{[g]}$$

It should be noted that the D_2 values were calculated conservatively because they imply an intake of 10^{-4} of the inventory ⁶.

As radioactive waste usually involves several radionuclides, for practical reasons the usual formula is proposed: ⁷

$$\sum_i \frac{C_i}{C_{Li}} \leq 1,$$

where C_i is the activity concentration of i radionuclide present in the RW and C_{Li} its corresponding threshold.

6. Proposal for the case of RW

Based on the reasons explained above the following proposal is made for RW. They are formulated for conditioned waste, i.e. for waste packages, though the basic concepts may be also applied for non-conditioned waste, such as waste in bulk or stored in tanks (e.g. spent resins stored in tanks in a nuclear power reactor)

- a) If an unshielded Radioactive Waste package has a dose rate at a distance of 1 m lower than 10 mGy/h and an activity concentration lower than $10^{-2} D_2/g$, the RW should not be considered a dangerous source and the level of security should be category IV of Nuclear and Radiation Related Threats [3] which represents the minimum level of threat.
- b) When one of these values is exceeded, that waste package should be categorized in function of the higher of the following ratios:
 - The ratio between the dose rate at 1 meter of the waste package and the limiting dose rate;

maximum intake of about 100 mg may be inferred from reference [8] and is limited to particular cases. A value of 10 mg is more realistic and, certainly appropriate when dealing with radioactive waste.

⁶ A more realistic fraction of 10^{-6} ; also considered in inhalation scenarios [5-9-10] seems to be more appropriate in the case of RW and values of inventory 100 times higher than D_2 could be derived.

⁷ Actually, this is a case where the usual formula is not applicable because deterministic effects are considered and not all radionuclides have the same target organ or tissue, but this approach is practical and conservative.

- The ratio between the actual activity concentration of the waste and the limiting concentration of activity.
- c) The accumulation of RW packages in a store bay should not imply an increase of the level of security.

As noted above, for disused sealed sources D values are directly applicable but considering the actual activity of the radioactive source instead of its activity at the time of manufacturing as recommended in [4].

7. Conclusions

The low activity concentration of RW, particularly of VLLW, LLW and ILW, makes inappropriate the direct application of the D Values. However, based on the same security principles, in particular the qualitative definition of a dangerous source, and using the same scenarios, it is possible to define generic security levels for RW.

It should be taken into account that the approach proposed might be modified in function of particular characteristics of the waste. For instance, RW included in a glass matrix may deserve a lower security level than the one that may result from the application of this approach.

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Nuclear Power Reactor Wastes. A Regulatory Overview of the Current Situation in Argentina

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Nuclear Power Reactor Wastes

A Regulatory Overview of the Current Situation in Argentina

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Abstract Since the middle of the 1980's up to the year 2005 the Argentinean nuclear power program was stagnated. Entering in 21st century the situation of the nuclear program had an inflection point as a consequence of the rise of the oil prices, the deep reduction of indigenous resources of natural gas and the increase of electricity demand that resulted in the growing of economical activity since 2001. At present the construction of Atucha II, the 3rd nuclear power reactor, has been restarted and negotiations for a 4th nuclear power reactor are in course. The only authorized site for final disposal of low level solid radioactive waste (AGE) suspended its operation at the end of the '90s and a new schedule for decision and commissioning of radioactive waste facilities were established by National Commission of Atomic Energy (CNEA). Nowadays, in the country, the radioactive waste (RW) from NPPs are stored in the same site where they are generated and it is not planned to have in operation a disposal facility before the second half of the next decade. So, the main technical regulatory concern during this period will be to assure that the storage of the radioactive waste is safe, the characterization of RW performed by Operators is appropriate and a reasonable recording system is implemented.

1. Introduction

The management of spent fuel (SF) and the disposal of high level radioactive waste (HLW) remain a key challenge for the nuclear power industry. Experts agree that the geological disposal of HLW is safe and technologically feasible. For the most advanced projects, disposal sites have been chosen and pre-construction work is under way. Even so, it will be more than a decade until the first of such facilities would be operational. In the meantime, the trend has been to construct and use aboveground interim storage facilities and many countries are exploring the feasibility of interim storage for 100 years or more. Some progress has been made in areas related to the disposal of particular types of low and intermediate level waste (LLW, ILW) [1]. In this paper the Argentine experience to resolve the safety management of nuclear power radioactive waste is discussed.

2. Situation in Argentina up to the middle of 1980's

In the last three decades, several international oil crises took place and some lessons were learned from them. The first one was that the period of cheap and abundant energy is over. The second was that the source of energy from oil was on the way of extinction. And finally, that the countries that don't have available energy resources are vulnerable. All that gave rise to the importance of nuclear energy in the world nowadays [2].

Until 1985, the Argentinean Nuclear Program (ANP) was an expanding and developmental program. One of the primary objectives of the ANP was the generation of nucleoelectric power by a complete fuel cycle of natural uranium with plutonium recycling. At that time, there were two NPP in operation, Atucha I of 330 MW(e) was operating since 1974 and Embalse of 600 MW(e) was operating since 1983, and a third NPP (Atucha II NPP of 745 MW(e)) that was under construction. Besides, in the country were underway the extraction, concentration and purification of uranium, the nuclear fuel manufacturing and the construction of a spent fuel reprocessing facility. Also research and development activities in uranium enrichment and the radioisotopes production were performed [3].

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During this period a Near Surface Disposal System for low level solid waste (AGE) was constructed. This system comprises two facilities commonly called trenches. The first one was commissioned in 1974 and in 1988 was completely closed. The second one was commissioned in 1989 [4].

At the end of the 1970s the Argentine authorities decided to explore alternatives for disposing of HLW that would eventually be generated by the ANP. The Argentinean authorities decided to explore potential solutions for the technological problems connected with the elimination of HLW. Feasibility studies aimed at characterizing a site for the final disposal of HLW were conducted during the 1980's but they were also suspended as result, inter alia, of postponing until the year 2030 the decision regarding reprocessing spent fuel [5-6].

3. The transitional period

Since the middle of the 1980's up to the year 2005 the Argentinean nuclear power program was stagnated. Several factors contributed to this situation, such as the drop of oil prices, new pit gas in exploitation and the economical and political crises that affected the development of the country and, therefore, reduced the electricity demand.

4. Current situation in Argentina

Entering in 21st century the previous situation of nuclear program had an inflection point as a consequence of the rise of the oil prices, the deep reduction of indigenous resources of natural gas and the increase of electricity demand from the raising of economical activity since 2001. At present the construction of Atucha II, the 3rd nuclear power reactor, has been restarted and negotiations for a 4th nuclear power reactor are in course.

The policy for long term storage and final disposal of radioactive waste had also changed in this period. The only authorized site for final disposal of low level solid radioactive waste (AGE) suspended its operation at the end of the '90s and a new schedule for decision and commissioning of radioactive waste facilities were established by CNEA. In 1997, by Act 25081, CNEA was appointed as the governmental organization responsible for the management of all radioactive waste generated in nuclear activities in the country [7].

In the meantime, all radioactive wastes produced by NPPs nowadays in Argentina are maintained in long-term interim storage facilities.

5. Perspectives and regulatory action

The Nuclear Regulatory Authority of Argentina (ARN) plays a twofold role:

1. It has to follow up, as the authority responsible for the licensing, the initiation of studies aimed at defining potential sites for the disposal of LLW and ILW in the country, as well as sites for the disposal of HLW.
2. It has to assure that the RW that is being generated at NPPs is well characterized and that appropriate records are kept at the facility level.

5.1 Activities foreseen

Related to the operation of Nuclear Power Plants (NPP) in our country, the interruption of storage and final disposal of LLW at the AGE forced the long-term interim storage of such waste at each facility. Besides, the ILW are being stored at the Nuclear Power facilities since the beginning of their operation.

Based on the results of the task carried out by an *Ad-Hoc* Regulatory working group during the year 2006 until mid 2007, it was concluded that the system to characterize RW at each NPP had to be improved and implemented as soon as possible.

The ARN required the facilities that at least, the following radionuclides present shall be identified in the RW characterization process.

C-14	Ni-59	Nb-94	Tc-99	I-129
Pu-238	Pu-239	Pu-240	Pu-241	Am-241
Cm-242	Cm-243	Cm-244	Np-237	H-3
Co-60	Ni-63	Sr-90	Cs-137	Cl-36

It should be noted that for structural radioactive waste, other radionuclides could be taken into account.

Additionally, it was also requested that the record keeping system should preserve the information for at least 10 years after the decommissioning of the facility. A copy of the record keeping system shall be transferred to the waste management entity together with the RW.

The record keeping system, shall include at least, the following information:

- ✓ The methodology used for the determination of activity concentration of each radionuclide (direct measurement, scaling factor method, computer codes, etc.)
- ✓ The radionuclides inventory of RW
- ✓ The physical form of RW
- ✓ The chemical composition of RW
- ✓ In situations where the RW are conditioned, the characteristics of the conditioning process and package shall be informed
- ✓ Identification of the emplacement of the RW

The system used for record keeping shall satisfy the following conditions:

- ✓ Capability of storing, keeping and preserving the information during a long time;
- ✓ Physically and chemically stable, so as the legibility remains in the long time;
- ✓ Easily copied or transferred to other system without losing the information;
- ✓ Resistance to alterations by non-authorized persons

6. Conclusions

Nowadays, in the country, the RW from NPPs are stored in the same site where they are generated and it is not planned to have in operation a disposal facility before the second half of the next decade. So, the main technical regulatory concern during this period will be to assure that the storage of the radioactive waste is safe, the characterization performed by Operators is appropriate and a reasonable recording system is implemented. These requirements shall be accomplished as soon as possible and the RW inventory at the storage facilities should be reevaluated taking into account the regulatory criteria for characterization.

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Experimental System to Give Reproducible and Controllable Conditions for Low Dose and Low Dose-Rate Beta Irradiation in Vitro

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EXPERIMENTAL SYSTEM TO GIVE REPRODUCIBLE AND CONTROLLABLE CONDITIONS FOR LOW DOSE AND LOW DOSE-RATE BETA IRRADIATION IN VITRO

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Abstract. We have developed an experimental system to irradiate cells in vitro with beta emission radionuclides. The system is economic, reproducible, and controllable, and offers the possibility to irradiate cell cultures with different sources of beta emitters in solution and, changing the initial activity, it is possible to irradiate with different initial dose rate. The effectiveness of this system was evaluated by measuring the production of oxygen reactive species in human normal fibroblast.

1. Introduction

Radioisotopes that decay via beta emission are widely used in science and medicine, particularly in the field of oncology. PET imaging, which exploits the basic mechanism of beta plus decay or positron emission, is becoming increasingly important in cancer diagnosis, follow-up evaluation, and radiation therapy planning. Beta-emitting radiopharmaceuticals are finding wider applications in cancer treatment, such as radioimmunotherapy and bone-seeking radiopharmaceutical therapy. Beta-emitting radioisotopes have also been extensively used in vascular brachytherapy and other brachytherapy applications. Many radioisotopes that undergo beta decay yield excited daughter nuclei, which produce gamma rays that are useful for both brachytherapy and teletherapy. [1]

The main advantage of beta-emitters is the relatively long path length in biological tissue (in the mm range), which is sufficient to irradiate cancer cells that do not have bound radiolabelled antibody efficiently and beta emitting source wires or seeds have been adopted in clinical practice of intravascular brachytherapy for coronary vessels.[2]

Many types of tumors overexpress cell surface antigens or receptors suitable as targets for radionuclide therapy. Different targeting agents have been suggested or are already being applied for such therapy. This therapy is currently employed for lymphomas [3,4] using radiolabelled antibodies and also for neuroendocrine [5-7] and pediatric tumors [8, 9] using radiolabelled somatostatin analogues and meta-iodobenzylguanidine (mIBG), respectively. In the majority of these cases, beta emitters such as ⁹⁰Y, ¹³¹I and ¹⁷⁷Lu have been applied.

The cell-killing capacity of low LET radiation, i.e. photons and electrons, is well known when applying at high dose rates, typically 0.5-2.0 Gy/min, as in external radiotherapy. A major difference is that the dose rate in radionuclide therapy is at least two orders of magnitude lower than in external radiotherapy [10-13]. Basic radiobiological studies have shown that low dose rates, in the range of 0.1-1.0 Gy/h, give a much lower biological effect (per dose unit) than high dose rates in the range 0.5-2.0 Gy/min [11,14,15]. But, it is also known, that an inverse dose rate effect exists in the range of 0.2- 0.4 Gy/h, that can give more cell killing than dose rates in the range 0.7-1.0 Gy/h [16]. In targeted radionuclide therapy is also

necessary to consider unwanted side effects on normal tissues.

The main objective of this work is to present the development of a simple device for irradiation of tissue culture with beta emitters, ^{32}P in solution, and his effectiveness for the study of production of reactive oxygen species in normal fibroblastic human cells at different dose and dose rate.

2. Materials and methods

2.1 Cells

The cells used were primary cultures of normal human fibroblast. They were grown in Dulbecco medium supplemented with 10% fetal calf serum, 2 mM glutamine, 100 ug/ml , streptomycin, 100 U/ml penicillin, all components from Gibco. The cells under irradiation were normally grown at 37°C with 5% carbon dioxide.

2.2 Irradiation system and dosimetry

The absorbed dose in tissue culture flasks was calculated applying MCNPX 2.5f Monte Carlo code coupled to photon and electron cross sections from ENDF/B-VI library [17]. Monte Carlo simulation was made following detailed experimental layout. In our model a thin water layer located in the bottom plane of each culture flask simulated cells. As absorbed dose rate decreases with time, code was used to calculate initial value, and total absorbed dose was obtained by time integration.

The cells were grown in the central area of the flasks were the initial average dose rate was $0,109 \text{ mGy/h} / \mu\text{Ci}$. Fig. 1 and 2 represents the schematic conditions of irradiation and computational model

Figure 1. Schematic representation of used irradiation conditions.

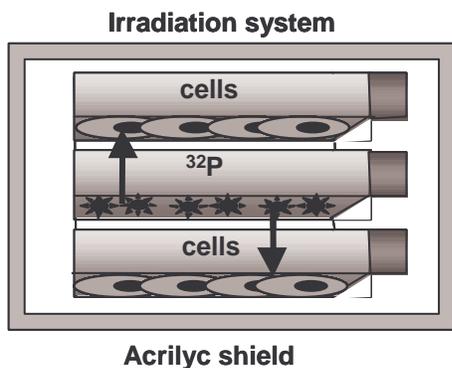
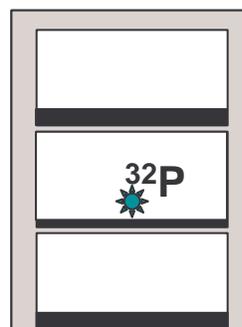


Figure 2. Computational model



2.3 Intracellular Reactive Oxygen Species (ROS) measurement

The intracellular production of ROS was evaluated by fluorescence microscopy, using an oxidation-dependent fluorogenic dye: 2'7'-dichlorofluorescein diacetate (DCFH-DA). This dye is absorbed by living cells and rapidly hydrolyzed by intracellular esterases to nonfluorescent DCFH [18]. In the presence of ROS, DCFH is oxidized to fluorescent dichlorofluorescein (DCF) [19], and the emitted fluorescence is directly proportional to the concentration of ROS. Control and irradiated cells were washed three times with PBS and incubated with DCFH-DA at a final concentration of 5 μM in PBS for 30 min at 37°C . Excess label was removed by washing with PBS prior to microscopy.

2.4 Results and conclusions

The effectiveness of this system was evaluated by the measuring the production of ROS in human normal fibroblast at 3 h and 24 h of exposure to beta emitters by fluorescence microscopy with the dichlorofluorescein probe. The initial average dose rate was 0,109 mGy/h / μ Ci and the final doses were 0,32 Gy, and 2,6 Gy respectively for the total activity. At both dose levels the average fluorescence intensity was higher than in the control.

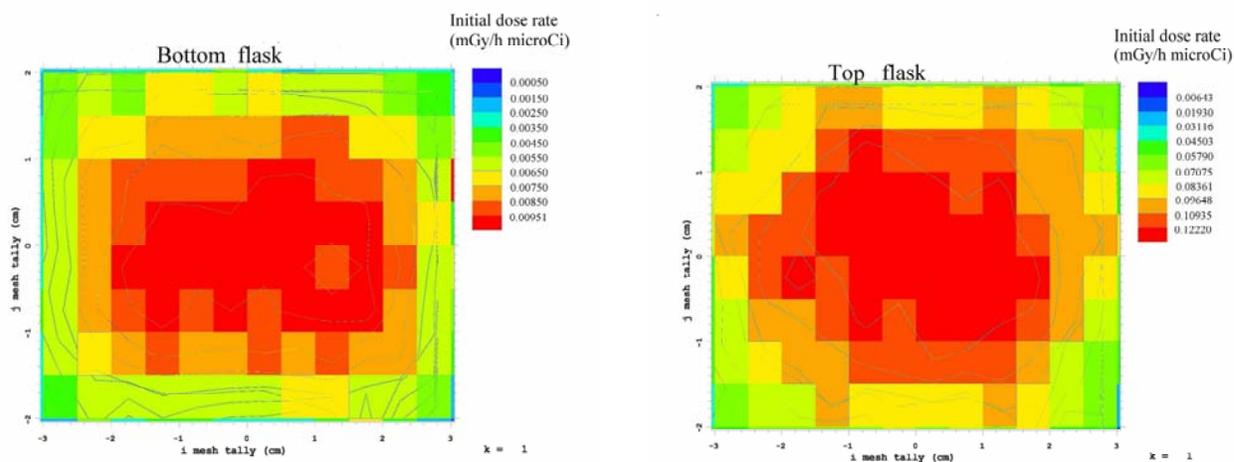


Figure 3. Initial dose rate distribution at top and bottom flasks

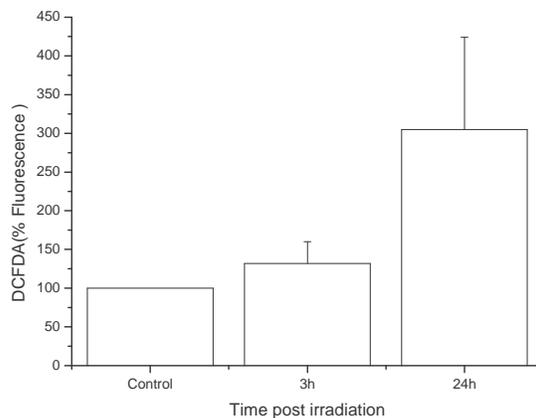


Fig 4 ROS induction by exposure to irradiation device for 3h and 24 h , measured by DCFH-DA fluorescence. The results are reported as a mean \pm SD

The system is economic, reproducible, and controllable, and offers the possibility to irradiate cell cultures with different sources of beta emitters in solution and, changing the initial activity, it is possible to irradiate with different initial dose rate.

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Probabilistic Safety Assessment (PSA) of the Radiotherapy Treatment Process with an Electron Linear Accelerator (LINAC) for Medical Uses

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Probabilistic Safety Assessment (PSA) of the radiotherapy treatment process with an Electron Linear Accelerator (LINAC) for Medical Uses.

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Abstract. This paper presents the results of the Probabilistic Safety Assessment (PSA) to the radiotherapy treatment process with an Electron Linear Accelerator (LINAC) for Medical Uses, which was conducted in the framework of the Extra budgetary Programme on Nuclear and Radiological Safety in Iberian-America. The PSA tools were used to evaluate occupational, public and medical exposures during treatment. The study focused on the radiological protection of patients. Equipment Failure Modes and Human Errors were evaluated for each system and treatment phase by FMEA. It was aimed at obtaining an exhaustive list of deviations with a reasonable probability of occurrence and which might produce significant adverse outcomes. Separate events trees were constructed for each initiating event group. Each event tree had a different structure since the initiating events were grouped according to mitigation requirements. Fault tree models were constructed for each top event. The fault trees were developed up to the level of components. In addition to hardware faults, the fault trees included human errors associated with the response to accidents, and human errors associated with the treatment. Each accident sequence was quantified. The combination of the initiating event and top events through one fault tree was the method used to analyse the accident sequences. After combining the appropriate models, a Boolean reduction was conducted by computer software to produce sequence cut sets. Several findings were analysed concerning the treatment process and the study proposed safety recommendations to avoid them.

KEYWORDS: *PSA; Risk, Radiotherapy; Safety.*

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1. Background

Reports of accidents in Radiotherapy have shown learned lessons about problems that have occurred and propose safety measures to avoid them [1-5]. Unfortunately these measures do not avoid completely other accidents, which can be originated by other root causes as is shown by experience. It is necessary, therefore, to find proactive methods and tools to analyze the vulnerabilities of the facilities and adopt measures that could avoid radiological accidental exposures.

Probabilistic Safety Assessment (PSA) is a proactive tool that has been successfully used in aeronautics, and nuclear and petrochemical industries. It can evaluate safety in an exhaustive and structured way, combining assessment of the effects of equipment faults, procedures and human errors and providing an insight into the strengths and vulnerabilities of the process being studied, the dominant contributors to the overall risk and the options to reduce it.

For these reasons, the Iberian-American Forum of Nuclear and Radiation Safety Regulatory Organizations, as part of its effort to promote the use of prospective safety analysis in radiotherapy, has realized a Project on Probabilistic Safety Assessment (PSA) during the radiotherapy treatment process with an Electron Linear Accelerator (LINAC) for Medical Uses (PSA-LINAC). That PSA analyzed the potential exposures during treatment originated by equipment failures or human errors during the different steps of the process.

This paper summarizes some of the main outcomes of that PSA Project.

2. Purpose and scope

The purpose of this Project was to conduct the safety analysis of the radiotherapy treatment with LINAC applying Probabilistic Safety Assessments tools for identifying the equipment failures or human errors that could lead to accidental exposures; as well as for assigning priorities for the regulatory control, prevention and mitigation in radiotherapy.

The study was focused in the analysis of the radiotherapy treatment process with a linear accelerator using as a reference a radiotherapy model established considering typical practices of the countries represented in the Project as well as practices that could be of interest for the present analysis. In this scope the analysis of some tasks that are external to the treatment (i.e. commissioning, calibrations or maintenance) was not included. It was assumed that they are performed successfully. It was so to focus efforts on those accidental exposures with less information from lessons learned out of documented accidents and with less perception about their likelihood. Issues not considered in the scope of this project could be dealt with by further PSA through other endeavours.

The PSA model included the human actions of the radiotherapy treatment team described in the project's scope, excluding those human actions that are part of the medical decision making process (for example dose prescription); because it was assumed the decisions were right, in conformance with the medical purpose.

3. Methods

The procedure for Probabilistic Safety Assessment [6] involves three fundamental tasks: the identification and definition of initiating events that may trigger accident sequences, the delineation of these sequences and the calculation of the accidental exposure frequency from each sequence.

The identification of the initiating events was performed using failure mode and effects analysis (FMEA) that is a standard method for identifying potential failures of an item of equipment, a system or a process and for analyzing the resulting effects [7]. As noted above, the FMEA method for identifying initiating events was applied to both, the hardware failure and the human actions required during the treatment process as considered in the scope of the Project.

The initiating events that were identified through the FMEA were grouped with the purpose of facilitating their use and treatment according to the following aspects:

- Similarity of safety barriers that avoid or mitigate the potential consequence
- Similarity of accidental exposures that the initiating events can yield.
- Possibility for modelling accidental sequences by a single event tree.

Each group of initiating events was treated as a single initiating event, and modelled as a logical fault trees, in a way such that no significant information resulting from FMEA was missed. Finally, a list of initiating events leading to potential exposures was obtained.

Events trees were used to delineate the possible accident sequences resulting from failures of relevant safety barriers currently implemented to avoid the progress of the initiating events that could lead to the accidental exposures.

Once this qualitative process was concluded the accident sequences were quantified to determine their annual frequency. The frequency of each accident sequence was calculated as the product of the initiating event frequency times the probability of failure of the barriers that are expected to act during the evolution of such sequence. Where no data was available for direct estimation of such probability, fault trees were used to model it, graphically and through logic gates, and combine the equipment failures and human errors that are root causes for the failure of the relevant barriers.

As a result of this quantification, the minimal cut sets were obtained (i.e. the minimum combination of component failures and human errors which produce an accident sequence). Once the event sequences had been identified and quantified, the most safety significant events were determined through importance analyses.

Generic Data Bases from several sources [8-11] were used to estimate the reliability of equipment as it is typically recommended for topical PSA that are applied for the first time; due to the low statistical significance of specific data on reliability of equipment and human errors in radiotherapy. For human error probabilities, screening values were used, i.e. conservative values which allow filtering the most important human actions, focusing efforts on them in further detailed analysis. These allow carrying out relative analysis from the absolute results obtained, since the whole quantification was done using the same type of data.

3.1 Definition of undesired events

The undesired events for this study are defined as accidental radiological exposures during treatment with LINAC which respond to the criteria indicated bellow. These criteria are based on the experience of several studies and publications [12-20] and on the consensus of the experts who participated in this PSA Project:

Criterion No.1: Group of people that receive accidental exposures

- a. *Workers (Z1)*: Any accidental exposure of oncologist, physicist, dosimetrist, therapist, nurses and paramedic of the radiotherapy service. Accidental exposures of biomedical engineer or equipment manufacturer engineer during setup, and maintenance of LINAC are not included.
- b. *Members of public (Z2)*: Any accidental exposure of member of public during the treatment process due to failure of safety systems and procedures established to avoid that exposure, for example inadvertent entrance of a patient's comforter to the treatment room. Irradiation of member of public due to poor shielding is not included. Amongst members of public were

included; patient's comforters, hospital workers who are not included in the radiotherapy practice, service and non-specialised maintenance workers, and visitors.

- c. *Patients (Z3)*: Any accidental exposure of patients, which is a deviation in more than 10% of the total prescribed dose. This misadministration may constitute an error in treatment delivery by over-dosage or under-dosage to the target volume; radiation dose to normal tissue outside intended treatment volume, not irradiated portion to intended target volume, or inhomogeneous dose to intended target volume.

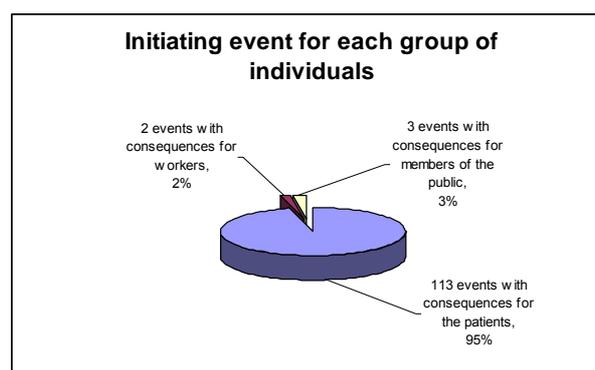
Criterion No. 2. Effect of misadministration on patient's treatment and number of affected patients [15]:

- *Individual Episodic (Z3A)*: Misadministration which affects a single treatment of one patient (lower than $\pm 10\%$ of the whole treatment). It can be recovered during the whole treatment and is not considered an accidental exposure.
- *Programmatic (Z3B)*: Misadministration which is higher than $\pm 10\%$ of the whole treatment of a patient.
- *Systematic (Z3C)*: Misadministration which can affect all patients treated on a specific service and it is higher than $\pm 10\%$ of the whole treatment.
- *Collective Episodic (Z3D)*: Misadministration which affects all treatments on a specific service, but it can be recovered during the whole treatment and is not an accidental exposure (lower than $\pm 10\%$ of the whole treatment).

4. Results.

During FMEA 453 failure modes or errors were identified which potentially might cause the undesired consequences. These were grouped into 118 initiating events. Likewise, 259 failure modes and human errors which can lead to the failure of the safety barriers were identified. Out of the 118 initiating events, 113 might lead to consequences for the patients, 2 for workers and 3 for members of the public, see Fig. 1.

Figure 1: Initiating event for each group of individuals.



The frequencies of the initiating events were determined as the product of their probability of occurrence and the annual frequency of the task where such initiating event could originate. The probability of occurrence was obtained by means of fault trees using generic data as noted above. The tasks' frequencies were obtained considering the average of the reference radiotherapy services.

In the PSA’s models 120 barriers were considered. Success/failure assessments of the barriers allowed identification of 434 accident sequences, out of which 115 can lead to systematic accidental exposures, 143 can lead to programmatic accidental exposures, 2 can lead to worker exposures and 3 can lead to exposures of members of the public. The remaining sequences represent misadministration that can be recovered during the progress of treatment and thus are not considered as accidental exposures within the scope of the study.

The results of the quantification of frequencies of accidental exposures are shown in Fig. 2- Fig. 5.

Fig. 2 shows the frequencies for patients accidental exposures (Z3) are dominant compared to the accidental exposures for workers (Z1) or members of the public (Z2).

Fig. 3 shows the quantification results referred to the type of misadministration.

Figure 2: Contributions of accidental exposures by groups of individuals with respect to the total annual frequency of potential exposures during treatment with LINAC

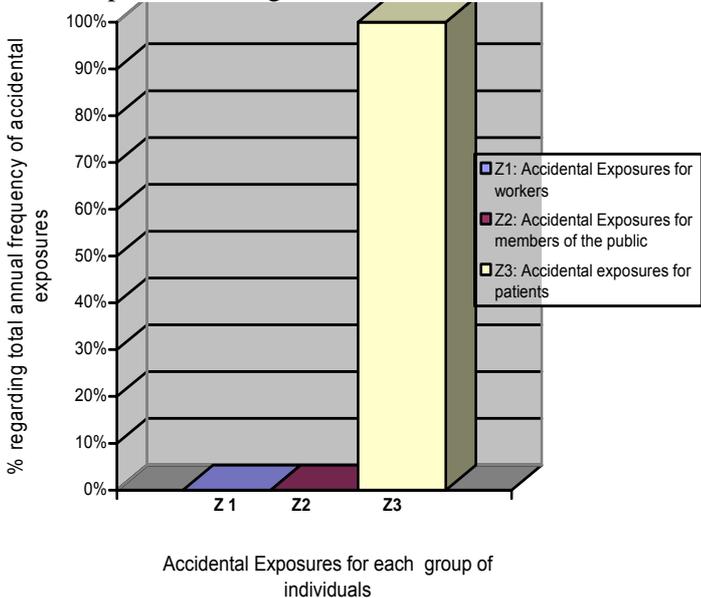
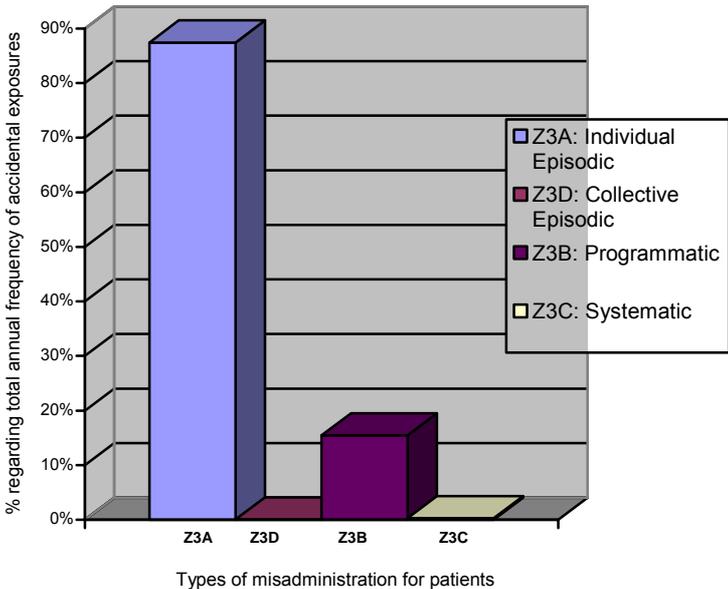


Figure 3: Contributions for different types of misadministration.



The results of the quantification of the accident sequences for programmatic exposures in the different stages of the treatment process are grouped in Fig 4. The results for systematic exposures are grouped in Fig 5.

Figure 4: Results of the quantification by treatment stage for programmatic accidental exposures of patients

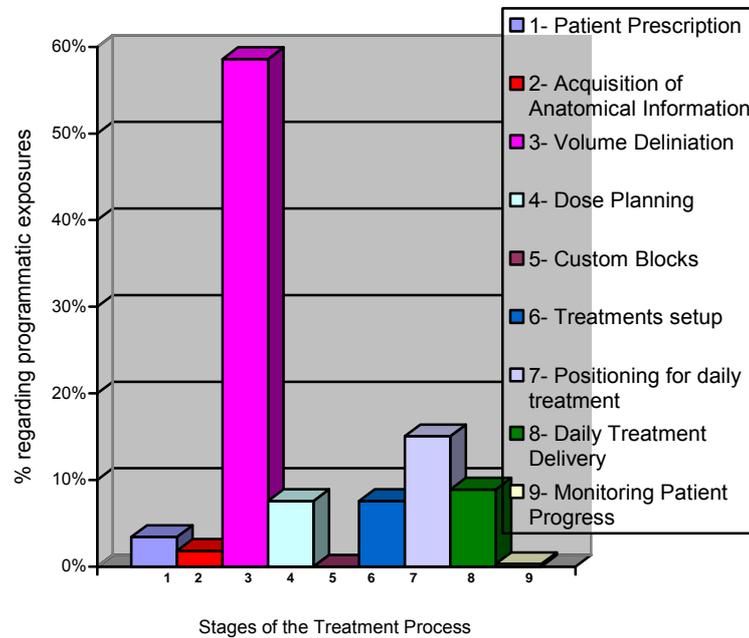
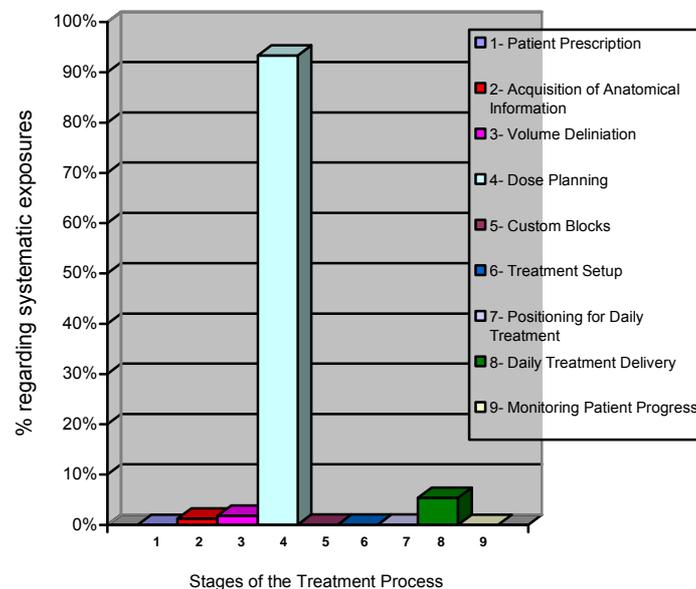


Figure 5: Results of the quantification by treatment stage for systematic accidental exposures of patients



5. Discussion

The 25 most probable types of accidental exposures involve patients (higher than 0.1 % of the annual total frequency of accidental exposures). None of the 25 is associated to initiating events triggered by equipment faults. Out of those 25; 24 are programmatic exposures.

The biggest contribution on the patient is episodic misadministration (87.5%). This type of misadministration is not accidental because it can be compensated during whole treatment progress.

Programmatic misadministration (Z3B) represents a 15.46 % of annual total frequency of accidental exposures. This type of misadministration is considered an accident because it affects patient's treatment in more than 10% of the whole treatment

The PSA study shows that *21 different event sequences are responsible for 90% of the potentially programmatic accidental exposures (Z3B). Three safety measures were identified that can avoid 55% of severe accidental exposures involving the whole course treatment of a single patient (programmatic).* These measures are the following:

- clinical evaluation by the radiation oncologist;
- in vivo dosimetry using reliable, calibrated detectors;
- approval of the treatment plan at a discussion/meeting of radiation oncologist and physicist.

On the other hand, the PSA shows that as few as *nine different event sequences are responsible for 90% of the potentially systematic accidental exposures (Z3C). Three safety measures can avoid 77% of the systematic or catastrophic accidental exposures.* These three measures are:

- Periodic quality control of the PC, digitizer, revalidation of the external beam (i.e. to check the constancy of external beam dose calculations to safeguard against inadvertent alteration or corruption [15]), transfer of the treatment plan;
- Validation after any modification of the TPS;
- Analysis of any change in the procedure for the use of the TPS.

The PSA study confirmed the need for proper commissioning of TPS in accordance with well proven protocols. This measure, together with regular quality control, would reduce by a factor of 21 the risk of catastrophic consequences; in particular, validation procedures for any change in the mode of use of the TPS can avoid catastrophic events similar to the Panama accidental exposure [4].

A number of errors relate to unclear delineation of target volumes and bear a significant contribution to accidental programmatic exposure. It is recommended: 1) to use a color code for those volumes and to make it mandatory in the radiotherapy department; 2) to include in TPS acceptance tests a verification of compliance with ICRU 62 [16] in connection with terminology; and 3) to incorporate into the design of the TPS interlocks and warnings to restrict manipulation of treatment volumes to alert staff on the potential omission of secondary treatment volumes.

The following safety measures have a preventive effect on a large number of initiating event sequences: 1) portal imaging at the initial session and periodically thereafter; 2) dosimetric tests; and 3) interlocks of the beam monitoring system. Absence of such safety measures increases the risk of initiating events where they apply by factors of 90, 30 and 6, respectively.

Independent review of the TPS calculation would substantially reduce the risk of accidental exposure. Absence of this safety measure increases the risk by a factor of 10.

The 'record and verify' system of medical accelerators drastically reduces the risk of nine initiating events related to daily treatment session delivery. Absence of this system increases the risk by a factor of 75, according to the computations of probabilities made in this research. New equipment should, therefore, include record and verify systems.

The presence of two technologists during treatment preparation and delivery is very important. Failure to comply with this good practice increases the risk of accidental exposure by a factor of 10. At least one of the two technologists should be the same during the whole course of treatment, from the initial setup until the end of treatment.

Conclusions

The PSA identified potential causes for accidents during treatment delivery and gave priorities for their attention regarding relevance and contribution to risk of accidental exposures. The study shows that PSA is an effective tool for evaluating and improving the safety of the radiotherapy treatment, complementing other traditional methods for evaluation of the radiological protection of patients. Also has been proved to be an excellent tool to improve the regulations and the control inspections.

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Radiation Safety Assessment of Cobalt 60 External Beam Radiotherapy using the Risk-Matrix Method

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RADIATION SAFETY ASSESSMENT OF COBALT 60 EXTERNAL BEAM RADIOTHERAPY USING THE RISK-MATRIX METHOD.

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Abstract. External beam radiotherapy is the only practice in which humans are placed directly in a radiation beam with the intention to deliver a very high dose. This is why safety in radiotherapy is very critical, and is a matter of interest to both radiotherapy departments and regulatory bodies. Accidental exposures have occurred throughout the world, thus showing the need for systematic safety assessments, capable to identify preventive measures and to minimize consequences of accidental exposure. Risk-matrix is a systematic approach which combines the relevant event features to assess the overall risk of each particular event. Once an event sequence is identified, questions such as how frequent the event, how severe the potential consequences and how reliable the existing safety measures are answered in a risk-matrix table. The ultimate goal is to achieve that the overall risk for events with severe consequences should always be low or very low. In the present study, the risk-matrix method has been applied to an hypothetical radiotherapy department, which could be equivalent to an upper level hospital of the Ibero American region, in terms of safety checks and preventive measures. The application of the method has identified 76 event sequences and revealed that the hypothetical radiotherapy department is sufficiently protected (low risk) against them, including 23 event sequences with severe consequences. The method has revealed that the risk of these sequences could grow to high level if certain specific preventive measures were degraded with time. This study has identified these preventive measures, thus facilitating a rational allocation of resources in regular controls to detect any loss of reliability. The method has proven to have an important practical value and is affordable at hospital level. The elaborated risk-matrix can be easily adapted to local circumstances, in terms of existing controls and safety measures. This approach can help hospitals to identify vulnerable aspects and improvements required and regulatory bodies to improve the quality of their inspections.

KEYWORDS: *safety assessment; risk analysis methods; radiotherapy.*

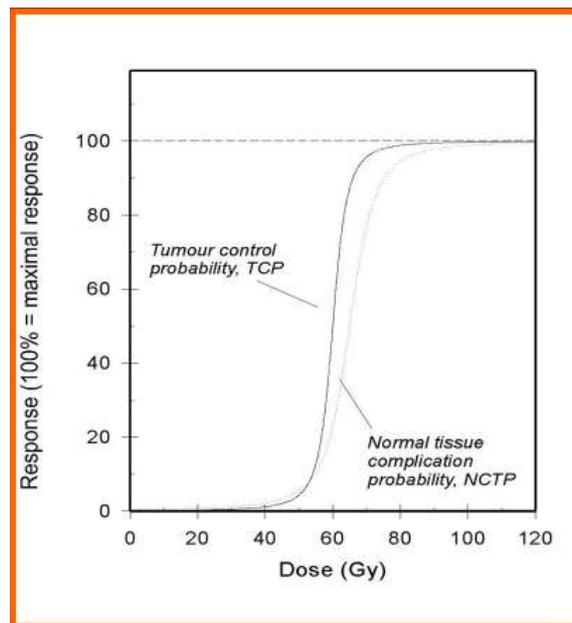
1. Introduction

Curative or palliative radiation therapy has three major concerns: efficacy, quality of life, and safety. From the point of view of radiation safety, radiotherapy is a very special application of radiation because it is the only one in which high doses of radiation is deliberately given to people (typically from 20 to 80 Gy).

Errors in radiotherapy, both overdosage and underdosage can be lethal or very severe. Figure 1 shows the principle of radiotherapy: the TCP curve represents the probability of tumour control and the NTCP curve shows the probability of normal tissue complications as a function of radiation dose. At low doses TCP and NTCP vary slowly with an increase in dose. From a given dose value both TCP and NTCP increase steeply but NTCP increases somewhat slower. For an extremely high dose both curves reach the value of nearly P=1 or 100%, i.e., both tumour and part of normal tissue are completely destroyed.

From this picture, a therapeutic window becomes apparent. At a given tumour dose (in this example, around 60 Gy) a probability of 80% for tumour control can be achieved with 20% normal tissue complication probability. Figure 1 also shows that the window is usually very narrow, hence the importance of accuracy and precision in dose delivery. This point is also crucial to safety, because equipment faults and human errors in the treatment process causing a 25% higher dose, for example, lead to a substantial increase of life threatening normal tissue complications. On the other hand, a 25% lower dose can cause a drastic decrease in tumour control probability.

Figure 1: Curves showing the variation of tumour control probability (TCP) and normal tissue complications probability (NCTP) as a function of the dose [1].



In addition, from the prescription to the delivery of a radiotherapy treatment, a team of professionals from a number of disciplines is involved in a large number of steps. In the case of external beam radiotherapy, a treatment is usually delivered with 20 to 40 fractions or sessions, each requiring a large number of machine and patient parameters, all of them similar but different from one patient to the next. For all these features radiotherapy has been receiving special attention in safety standards, in particular the International Basic Safety Standards for the Protection against Ionizing Radiation and for the Safety of Radiation Sources (the BSS) [2], which establish requirements to investigate accidental medical exposure and to adopt corrective measures to avoid recurrence.

There is substantial amount of literature with detailed reports on major accidental exposure [3, 4, 5], and the lessons learned from them, which help identify measures to avoid similar accidental exposures in future. In spite of these efforts, accidental exposures continue to happen, some time showing new weaknesses, which were not identified by safety assessments traditionally performed for radiotherapy. In this situation more proactive approaches to risk analysis are gaining attention.

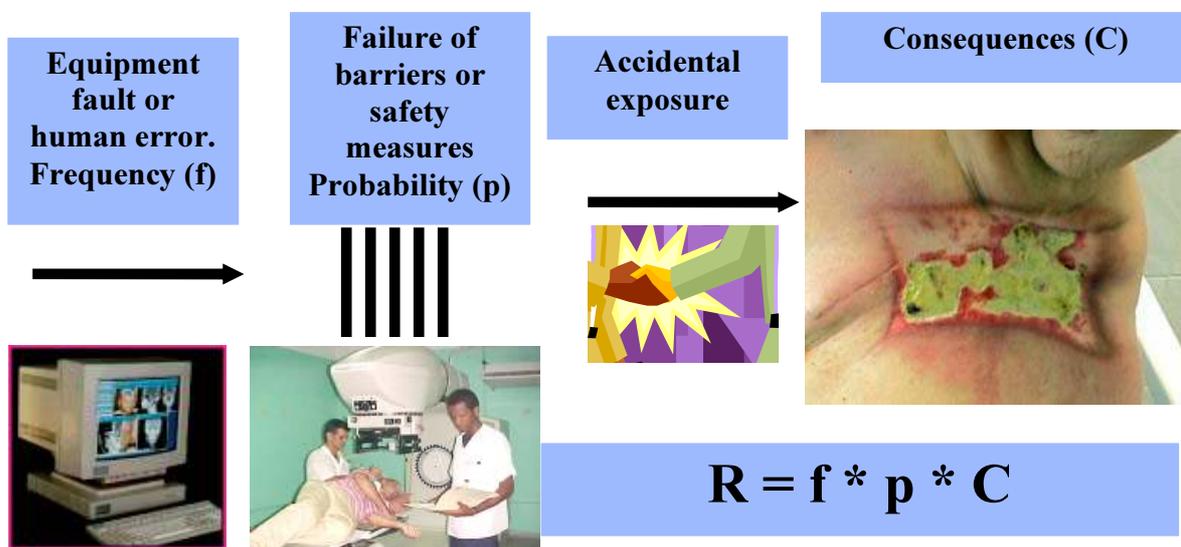
Probabilistic Safety Assessments (PSA) is a comprehensive and structured approach, which has been recently used to evaluate risk in radiotherapy. Results of the application of risk analysis methods have been published in [6]. However, the complexity of this method and the unavailability of human resources that can be devoted to this task in radiotherapy departments, have prevented its use on a wider scale. In this work a safety assessment of ^{60}Co external beam radiotherapy treatments using the risk matrix approach, as proposed in [7], has been carried out for a ^{60}Co treatment unit with pneumatic control source drawer.

2. Safety assessment of cobalt 60 external beam radiotherapy using risk-matrix method

2.1 Technical basis for the risk matrix method.

In order to understand the risk matrix method it is necessary to follow the event sequence leading to the accidental exposure. Figure 2 shows a simplified scheme of the event sequence.

Figure 2: Simplified scheme of an accidental sequence



A fault or error (initiating event), which can lead to an undesired outcome, occurs at a given frequency (f), depending on the type of fault or error. The problem can be detected and corrected if certain elements (interlocks, alerts or alarms, or double checks procedures) are in place in order to prevent the accidental exposure. These elements are called “barriers”. However, there is a certain probability of failure of these barriers in which case, the accidental exposure with its consequences (C) will be inevitable, affecting the patient, workers or the public.

The global parameter which characterizes a given accidental sequence is the risk or detriment¹ (R), which expresses the combined probability of harm.

As shown in figure 1,

$$R = f * p * C. \quad (1)$$

With the risk matrix method each of the variables of equation (1) is evaluated for each initiating event, assigning to f, p, C one of four levels previously established, and obtaining in each case one of the four levels for the resulting risk (R), as shown in Table 1.

¹ The risk is used in this context is broad as human or health risk, which includes the more specific term “detriment”, in this case from the accidental exposure, which is the result of two events: first, the occurrence of the accidental exposure and second the appearance of the associated harm. The associated harm to be considered when the end points are assessed includes death or serious localized harm from deterministic effects, cancer and hereditary harm due to stochastic effects.

Table 1: Complete risk matrix containing all combinations of the four levels of frequency of occurrence of the initiating even (f), the four levels of probability of failure of the set of safety measures (P) and the four levels of severity of consequences (C), if the initiating event results in an accidental exposure.

f _H	p _H	C _{VS}	R _{VH}
f _M	p _H	C _{VS}	R _{VH}
f _L	p _H	C _{VS}	R _H
f _{VL}	p _H	C _{VS}	R _H
f _H	p _M	C _{VS}	R _{VH}
f _M	p _M	C _{VS}	R _H
f _L	p _M	C _{VS}	R _H
f _{VL}	p _M	C _{VS}	R _H
f _H	p _L	C _{VS}	R _H
f _M	p _L	C _{VS}	R _H
f _L	p _L	C _{VS}	R _L
f _{VL}	p _L	C _{VS}	R _L
f _H	p _{VL}	C _{VS}	R _H
f _M	p _{VL}	C _{VS}	R _L
f _L	p _{VL}	C _{VS}	R _L
f _{VL}	p _{VL}	C _{VS}	R _L

f _H	p _H	C _S	R _{VH}
f _M	p _H	C _S	R _H
f _L	p _H	C _S	R _H
f _{VL}	p _H	C _S	R _H
f _H	p _M	C _S	R _H
f _M	p _M	C _S	R _H
f _L	p _M	C _S	R _H
f _{VL}	p _M	C _S	R _L
f _H	p _L	C _S	R _H
f _M	p _L	C _S	R _H
f _L	p _L	C _S	R _L
f _{VL}	p _L	C _S	R _L
f _H	p _{VL}	C _S	R _L
f _M	p _{VL}	C _S	R _L
f _L	p _{VL}	C _S	R _{VL}
f _{VL}	p _{VL}	C _S	R _{VL}

f _H	p _H	C _{MO}	R _H
f _M	p _H	C _{MO}	R _H
f _L	p _H	C _{MO}	R _L
f _{VL}	p _H	C _{MO}	R _L
f _H	p _M	C _{MO}	R _H
f _M	p _M	C _{MO}	R _L
f _L	p _M	C _{MO}	R _L
f _{VL}	p _M	C _{MO}	R _L
f _H	p _L	C _{MO}	R _L
f _M	p _L	C _{MO}	R _L
f _L	p _L	C _{MO}	R _L
f _{VL}	p _L	C _{MO}	R _L
f _H	p _{VL}	C _{MO}	R _L
f _M	p _{VL}	C _{MO}	R _L
f _L	p _{VL}	C _{MO}	R _{VL}
f _{VL}	p _{VL}	C _{MO}	R _{VL}

f _H	p _H	C _{mn}	R _L
f _M	p _H	C _{mn}	R _L
f _L	p _H	C _{mn}	R _L
f _{VL}	p _H	C _{mn}	R _L
f _H	p _M	C _{mn}	R _L
f _M	p _M	C _{mn}	R _L
f _L	p _M	C _{mn}	R _{VL}
f _{VL}	p _M	C _{mn}	R _{VL}
f _H	p _L	C _{mn}	R _{VL}
f _M	p _L	C _{mn}	R ^{VL}
f _L	p _L	C _{mn}	R _{VL}
f _{VL}	p _L	C _{mn}	R _{VL}
f _H	p _{VL}	C _{mn}	R _{VL}
f _M	p _{VL}	C _{mn}	R _{VL}
f _L	p _{VL}	C _{mn}	R _{VL}
f _{VL}	p _{VL}	C _{mn}	R _{VL}

2.2. Criteria to assign levels to frequency of initiating events, (f), probability of barrier failure (p) and consequences (C).

A key step is the assignment of levels to each of the independent variables of the risk matrix in (1). It is done starting from an expert judgment by various specialists participating in the assessment (physicians, physicists, dosimetrists and technologists). This method of assignment provides higher objectivity. Although the four levels of the independent variables are obtained in a qualitative manner, the use of quantitative criteria as reference in the assignments, justifies to call the risk matrix approach a “semi quantitative” method. In the following sections the quantitative reference criteria are given.

2.2.1 Frequency of occurrence of initiating events.

The frequency of initiating events (IE) can be obtained directly from the historical records of equipment faults and human errors existing at a given facility. However, there are no reliable records of faults and errors available in the region. The annual frequency, f , can be estimated as the probability of occurrence of the fault or error (P^{IE}), each time an action or task is performed, multiplied by the number of times that the task is done in a year (N^E). In [8, 9] values of probability of equipment faults and human error can be found, although they are not specific for radiotherapy, they can be used with acceptable approximation for the purposes of this study. The number of times that a given task or action is performed each year can be calculated from the radiotherapy department records (for example, from the annual number of patients, average number of sessions per patient, number of treatment fields per patient). In this way the frequency can be estimated from equation (2).

$$f = P^{IE} * N^E \quad (2)$$

The estimated value of the frequency of occurrence of each initiating event IE (f) is then compared with table 2, from which the frequency interval or level, required by the risk matrix, can be assigned.

Tabla 2: Annual frequency intervals of initiating events in number of times per year.

Qualitative frequency	Abbreviation	Annual number of events (considering 500 patients per year)
High	f_H	$f \geq 50$
Medium	f_M	$1 \leq f < 50$
Low	f_L	Between 1 per year and 5 in 100 years $0.05 \leq f < 1$
Very low	f_{VL}	Less than 5 every 100 years. $f < 0.05$

2.2.2 Potential consequences.

Assignment of consequence levels (C) is done assuming that the initiating event has occurred and all existing barriers have failed, i.e. the accidental exposure has occurred. Considering that consequences are different for patients than for workers and members of the public, two different tables are proposed.

2.2.2.1 Consequences for patients.

The following scale has been used

- 1- Catastrophic or very severe, (C_{VH}): accidental exposures affecting many patients and causing multiple deaths or irreversible handicap injuries.
- 2- Severe (C_S): accidental exposure causing death or irreversible handicap injuries to one single patient.
- 3- Moderate (C_{MO}): accidental exposures affecting a single patient for one session, with low deviation in the total dose.
- 4- Minor (C_{mn}): Loss of defence in depth, not causing any significant deviation in dose.

2.2.2.1 Consequences for workers and the public.

- 1- Catastrophic or very severe (C_{VS}): fatal or life threatening consequences, severe deterministic effects, irreversibly reducing the quality of life.
- 2- Severe (C_S): non life-threatening, reversible consequences, not affecting quality of life
- 3- Moderate (C_{MO}): anomalous exposure (not included in normal exposure), exceeding dose constraints and limits, below thresholds for deterministic effects
- 4- Minor (C_{mn}): Loss or degradation of defence in depth not causing any significant exposure

2.2.3 Failure probability of the set of barriers.

There are different kinds of barriers (interlocks, alarms, procedures) and their strength depends on many factors. When applying the risk matrix, failure probability for the set of barriers is conservatively estimated by taking only the number of barriers into consideration. All barriers are assigned the same individual probability, at the highest expected level.

After completing the risk matrix and screening off the event sequences of low and very low risk, a deeper probability analysis for the set of barriers is carried out in order to reassign the level in a more realistic manner.

The first, conservative, assignment, based on the number of barriers only, follows the rules below:

- 1- High (p_H): there is no safety barrier.
- 2- Medium (p_M): there is only one or two safety barrier in the set
- 3- Low (p_L): there are three safety barriers in the set
- 4- Very low (p_{VL}): There are four or more safety barriers in the set.

2.3. Step sequence in the application of the risk matrix method

2.3.1 List of initiating events.

An important warning is that any initiating event, which is not postulated, will not be analyzed and will remain as a potential event, not evaluated and therefore hidden. The first step is, therefore, to identify all equipment faults and human errors that may lead to one of the four levels of postulated consequences.

In this work the list of initiating events was taken from [10], considering that that study was done on a ^{60}Co unit of the same type and for the same hypothetical radiotherapy department of the Ibero American region. The list of initiating events was thoroughly reviewed and was also checked against the lessons learned from reported accidental exposures [2, 3, 4, 11], applying the principle that any event that has already occurred must be included in the analysis. As many as 76 initiating events from all stages of the treatment process were identified.

2.3.2 Potential consequences from the postulated initiating events.

The consequences from each initiating event is analyzed, by assuming first that after it has occurred there is no barriers in the way towards an accidental exposure. At this stage, possible elements that could reduce the consequences are considered (consequence reducers). These reducers are not like barriers, i.e., they can not avoid the consequences but can mitigate them.

In this work, the level of consequences was estimated by following expert judgment, considering the deviation in radiation dose from each initiating event and the characteristics of the accidental exposure (affecting one fraction, or the whole treatment or all patients). In this study there was no need to make calculations or simulations to confirm the magnitude of deviations in dose, but this is also an option that can be used if needed.

2.3.3 Frequency of the initiating events

First the number of times that an initiating event occurs in a given time is assessed, usually the annual frequency. It is important to use the records of equipment faults and human errors that may exist in the radiotherapy department. At this stage, frequency reducers are also identified. These elements act always before the initiating event.

For this work no reliable records were available to estimate the frequency, so the assessment was based on published data and on applying equation (2).

2.3.4 Existing safety barriers and probability assignment to the failure of the barrier set.

All existing barriers are identified, i.e., all technical or organizational measures established to avoid the consequences from the initiating events. Barriers usually belong to one of the following types:

1. Interlocks: technological devices responsible for a safety function, able to automatically detect an unsafe condition and remove it or stop the irradiation.
2. Alarms: audible or visual signals to alert people on the need to take action or make a decision; they require, therefore, human action.

3. Procedures: written instructions or standardized practices, to avoid or detect errors and deviations in the various tasks of a given process.

At this stage, only direct barriers are considered, i.e., those that when called upon by an initiating event acts to avoid the postulated consequences.

Barrier set failure (p) is analyzed by estimating the combined probability of failure of the whole set of barriers, relevant to the initiating event. As stated in 2.2.3, this estimation, at first, is very conservative since it only considers the number of barriers, without taking barrier strengths into consideration.

2.3.5. Resulting risk levels from the risk matrix

Once the independent variables (f , p , C) with their respective levels have been entered into table 1, the matrix provides the resulting risk level as very high, high, low or very low.

This process is done for each accidental sequence. It is important to understand that, as a result of the methodology, risk levels are obtained and no risk values. Two initiating events with the same risk levels do not necessarily have equal risk.

2.3.6. Result analysis.

At first, all event sequences whose resulting risk level falls into the “low” or “very low” category are screened off and a more detailed analysis focuses on the remaining event sequences, i.e., those identified by the matrix as level “high” or “very high” risk. In this analysis, all frequency and consequence reducers and barrier strengths are brought to consideration.

Sequences identified by the matrix as low or very low risk can be excluded from further analysis, considering the conservative features of the preliminary analysis and the risk management criteria shown in table 2 below.

Table 2: Risk management criteria

Risk interval	Risk acceptability	Actions
Very high (R_{VH})	Unacceptable	It would be sensible to stop the practice or the relevant treatments until measures to reduce the risk have been implemented
High (R_H)	Unacceptable when consequences are very severe (catastrophic) or severe	Immediate measures are required to reduce risk or the practice would need to be stopped
	Acceptable temporarily under certain conditions if the consequences are low or very low.	Measures of risk reduction need to be implemented without unnecessary delay
Low (R_L)	Acceptable from the cost-benefit point of view	Potential improvements should be explored but implementation should only proceed from cost-benefit view point.
Very low (R_{VL})	Negligible.	No action is required

In this work the analysis proceeds by asking the following questions:

A1-Are the existing barriers robust enough as to justify a reassignment of failure probability to a lower interval (for example from “high” to “medium” or “low”)?

A2-Can the frequency of the initiating event be reduced, using identified frequency reducers?

A3-Can consequences of the initiating event be reduced, using identified consequence reducers?

A4-Can additional barriers or measures be introduced to reduce the overall risk?

Basic reducers can be found in the following way:

- Reducers of the frequency of initiating events are usually associated to maintenance policies for equipment, and keep human errors as low as possible by selection and qualification of staff, moderating workload and improving the working environment to invite concentration and avoid frequent distraction and bad habits, i.e. a safety culture approach.
- Reducers of consequences of initiating events are those associated to quality control programmes and compliance with procedures for patient following up.
- Reducers of the probability of failure of barriers include the introduction of new barriers or strengthening existing ones, in order to make them more robust and reliable.

2.3.6. Drawing recommendations

Once new barriers, frequency and consequence reducers are identified, drawing recommendations follows. Recommendations should be tailored to help hospital administrators and managers to do a rational, risk-informed, prioritized allocation of human and material resources with a higher impact on the risk.

4. Conclusions

Risk matrix is a systematic simplified method, derived from the more complex probabilistic safety assessment (PSA) techniques, which can be easily applied by staff not necessarily specialized in PSA. This method, although it does not quantify risk with the accuracy of a PSA, helps identify priorities in a structured manner, without a more detailed quantitative study. The risk matrix method can be used by radiotherapy departments as well as by regulatory bodies.

The risk matrix approach has been applied to ⁶⁰Co external beam radiotherapy treatments and evaluation 76 event sequences of a hypothetical radiotherapy department, has been done. The study has corroborated that this hypothetical department is protected against “severe” and “very severe” consequences from initiating events identified as having an overall risk at “high” and “very high” level.

This hypothetical department has the highest level of safety that can be expected in the region, with a reasonable number of barriers and good safety practices. It should, therefore, not be assumed that all radiotherapy departments in the region are protected against all these initiating events. The preliminary application of the risk matrix, without the result analysis described in 2.3.6 (questions A1, A2, A3), identified 23 accident sequences with “severe” or “very severe” consequences, which were preliminary rated as “high” risk level. These risks could be reduced to low level, by considering the robustness of barriers and reducers. It is particularly important to monitor these barriers and reducers, since their failure can compromise life of one or multiple patients.

Five event sequences were identified as having “medium” level consequences associated to an overall “high” risk. No additional barriers were identified for these initiating events, because they occur just before daily treatment. It is essential, therefore, to implement reducers of frequency of the initiating event and supervise their effectiveness in a systematic manner.

In a more general sense, for all event sequences evaluated, supervision of effectiveness of barriers and reducers is essential so that effectiveness is maintained over time. Otherwise risks identified as low and very low may increase as well. External audits should address these elements specifically.

The study has revealed the need for additional barriers to reduce failure probability, in particular the following barriers should be considered:

- New patient planning conference as proposed by AAPM TG 40. These meetings contain a number of frequency reducers.
- In vivo dosimetry. Although other documents, such as AAPM TG 40; TECDOC-1151 [12, 13,] recommend them, it is not a generalized practice in countries of the Ibero American region.
- Two technologists per treatment machine and per shift, one of them verifying compliance with procedures.
- External audits before start clinical use of equipment and periodically during operation. It is important that audits include supervision of barriers and reducers as identified in this study.

The study has revealed that frequency and consequence reducers are very important to reduce risk, and in this sense the following reducers are mentioned:

- Training of radiation oncologists, medical physicists, dosimetrists and technologists, including the lessons from accidental exposure and the result of proactive study similar to this work.
- Standardized forms and protocols for prescribing, planning and reporting treatment information.
- Keeping workload to a moderate level.
- Programme of preventive maintenance.
- Implementing policies and measures to keep a working environment that stimulates work with attention, due thought, full knowledge and a proper sense of accountability (safety culture). This applies also to design (for example placing the control console on a place not readily accessible to other people).
- Having sufficient trays with fixed blocks for the whole treatment of a given patient.
- Weekly patient follow-up.
- Quality control of equipment parameters.

One of the difficulties detected by the risk matrix method was that of assigning frequencies to initiating events. This is due to scarce records on events that have occurred in the hospital. In this respect, hospital administrators and managers should create a non-punishing, stimulating atmosphere favouring reporting and recording these events.

When discharging regulatory duties of licensing, inspection, regulators have the possibility of including key aspects with impact on risk, identified in this study.

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Risk Analysis Methods: their Importance for Safety Assessment of Practices using Radiation

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RISK ANALYSIS METHODS: THEIR IMPORTANCE FOR SAFETY ASSESSMENT OF PRACTICES USING RADIATION

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Abstract. Radiation safety has been based for many years on verification of compliance with regulatory requirements, codes of practice and international standards, which can be considered prescriptive methods. Accident analyses have been published, lessons have been learned and safety assessments have incorporated the need to check whether a facility is ready to avoid accidents similar to the reported ones. These approaches can be also called “reactive methods”. They have in common the fundamental limitation of being restricted to reported experience, but do not take into account other potential events, which were never published or never happened, i.e. latent risks. Moreover, they focus on accident sequences with major consequences and low probability but may not pay enough attention to other sequences leading to lower, but still significant consequences with higher probability. More proactive approaches are, therefore, needed, to assess risk in radiation facilities.

They aim at identifying all potential equipment faults and human error, which can lead to predefined unwanted consequences and are based on the general risk equation: Risk = Probability of occurrence of an accidental sequence * magnitude of the consequences. In this work, a review is given of the experience obtained by the countries of the Ibero American Forum of Nuclear and Radiation Safety Regulatory Organizations, by applying proactive methods to radiotherapy practice. In particular, probabilistic safety assessment (PSA) used for external beam treatments with linear electron accelerators and two studies, on cobalt 60 therapy and brachytherapy using the risk-matrix approach are presented. The work has identified event sequences, their likelihood of occurrence, the consequences, the efficiency of interlocks and control checks and the global importance in terms of overall risk, to facilitate decision making and implementation of preventive measures. A comparison is presented of advantages and limitations of each method, in terms of feasibility of application in practice and of resources required. Finally, ways are proposed to extend this experience to other countries of the Latin American and other regions.

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KEYWORDS: *safety assessment; risk analysis methods; radiotherapy.*

1. Introduction

Safety in radiotherapy has received emphasis by international organizations and professional bodies worldwide. The International Basic Safety Standards for the Protection against Ionizing Radiation and for the Safety of Radiation Sources (the Basic Safety Standards, BSS) require that licensees shall perform a safety assessment of the radiation sources under their responsibility and specifies further that these assessments shall include, a systematic critical review of ... the ways in which structures, systems, components and procedures related to protection or safety might fail, ... and the consequences of such failures.

In spite of this requirement, more than 10 years after BSS were issued, systematic safety assessments are still scarce for radiotherapy, either because the tools for these assessments are not well developed or adapted for radiotherapy or because technical data to perform this assessment are not yet available. Due to the absence of systematic approaches, efforts have been restricted to apply the lessons learned from major accidental exposure. Basically, this approach is deterministic in nature. A scenario of major consequences in terms of radiation exposure, which has been learned evaluated from reported events, is postulated and measures are taken to avoiding reoccurrence or reducing their probability to a very low value. Consequently with this approach regulations require prevention of these scenarios by defense in depth, i.e. multiple barriers or multiple barriers or safety measures¹.

Assessments focused only on major accidental exposures may overlook others with significant, but less severe consequences, which due to their higher probability of occurrence and the absence of sufficient barriers, their combined risk may be higher than that of the major accidental exposures. This insight can only be obtained by more systematic approaches, such as the ones presented in this report.

Lessons from major accidental exposures reported in the open bibliography [2,3,4,5] point to the fact that the event sequences leading to them had not been anticipated and, as a result, no sufficient layers of defense were placed. In this report, a proactive approach to safety assessment is presented, which up to date has been used to a very limited extent in radiotherapy. The approach includes two methodologies for risk analysis, which have been adapted to radiotherapy.

This work has been carried out under the auspices of the Ibero American Forum of Nuclear and Radiation Safety Regulatory Agencies (the FORO)

2. Risk analysis methods

2.1. Approach for safety evaluation: Strengths and limitations.

Three methods are available to assess or verify the safety in facilities using radiation: prescriptive, reactive and proactive approaches. All three are useful but there are strengths and limitations for each of them.

2.1.1. Prescriptive approach

The prescriptive approach is used to verify compliance with a list of requirements given by regulations, design codes or standards, in which the results of historical evidence and of research and development are reflected. The nature of the method is to check whether pre-established requirements are satisfactorily met. The effectiveness of this method has been shown in practice over many years. However, this method can be supplemented and strengthened by other methods that go beyond compliance with requirements but proactively search for potential weaknesses.

2.1.2. Reactive approach

This approach is carried out by searching for solutions to the problems revealed by previous major accidental exposures, and are aimed at avoiding their reoccurrence. Classical methods of “root cause analysis” fall into this approach. This method has the advantage of easily sensitizing high-level administrators in hospitals who will be willing to provide the necessary resources to prevent them

Their main limitation is given by the fact that only well documented accidents can be analyzed, and these are usually the most important, catastrophic-type of events. Many other events that have occurred but were not adequately reported cannot be rigorously analyzed and lessons cannot be drawn and used.

More recently, information on near misses without consequences is collected using anonym reporting systems, such as the Radiation Oncology Safety Information System (ROSI), so that lessons can be learned without waiting for a major accidental exposure to happen.

¹ Defense in depth is defined in the International Basic Safety Standards as “the application of more than a single safety measure for a given safety objective so that the objective is achieved even if one of the protective measures fails”. In radiotherapy, examples of safety measures are an interlock of the beam monitoring system or double-check procedure.

2.1.3. Proactive approach

The proactive approach is based on a search for all potential failures and errors, estimating their risk and prioritizing efforts to avoid accidental exposure from those events leading to higher risk. The approach provides a kind of “made-to-measure” assessment for each particular facility. The main limitations of the method is that, in general, it is highly time-consuming and requires the involvement of a multidisciplinary group of specialists, being for this reason costly and likely to be unaffordable for small hospitals and clinics with limited human and material resources.

In this report emphasis is made on concrete examples of techniques to incorporate proactive approaches into safety assessments in radiotherapy, and efforts are made to make the benefit of proactive methods affordable to individual hospitals. This is not meant to replace reactive approaches, but rather both methods should be used and combined in an effective manner, in order to raise the technical level and accuracy of the safety assessments.

2.2. Common aspects.

Any risk analysis and risk reduction method should follow a common step-by-step process to systematically and thoroughly analyze a given practice. The steps are the followings:

1. Risk identification
2. Risk quantification
3. Result analysis and decision making.
4. Implementation of risk reduction measures

2.2.1. Identification of potential risks.

In this stage all possible equipment faults and human errors which may lead to undesirable, previously postulated consequences, are identified. In the case of radiotherapy, the undesirable consequences are those defined in the BSS, i.e. “any therapeutic treatment delivered to either the wrong patient or the wrong tissue, ..., or with a dose or dose fractionation differing substantially from the values prescribed ...”.

There are a number of tools for risk identification reported in the literature [6,7]. For this work, the technique called “failure mode and effects analysis” (FMEA) was chosen. FMEA is a standardized method to identify, in a systematic manner, potential faults of equipment, systems or processes and analyze the effects of these faults, with respect to a given undesired outcome. FMEA is carried out in three steps:

1. All possible failure modes and human errors are identified in every equipment and every task of the treatment process
2. When a failure mode is identified, its causes, consequences and existing defenses are analyzed and recorded in an organized way
3. This process is repeated for every fault of the equipment and for all pieces of the equipment of the system or treatment step
4. When the system or treatment step is completed the next one is started until the whole treatment process is done

FMEA proceeds as brain storming sessions in technical meetings. The analysis is performed at independent sessions for each of the systems included in the scope of the study, with participation of a multidisciplinary group. In the case of radiotherapy, multidisciplinary groups included a radiation oncologist, technologist, physicists and risk analysis specialist. FMEA results are documented in tabular form.

2.2.2. Risk quantification.

In this stage a risk estimation is made from the general equation of risk:

$$R = f \bullet C. \quad (1)$$

Where:

R: risk of the evaluated event sequence.

f: probability of occurrence of the event sequence leading to the accidental exposure

C: magnitude of the consequences associated to the occurrence of the accidental exposure

For this work, two quantification techniques were selected to be applied to the radiotherapy practice: the “probabilistic safety assessment” (PSA) and the “risk matrix” method.

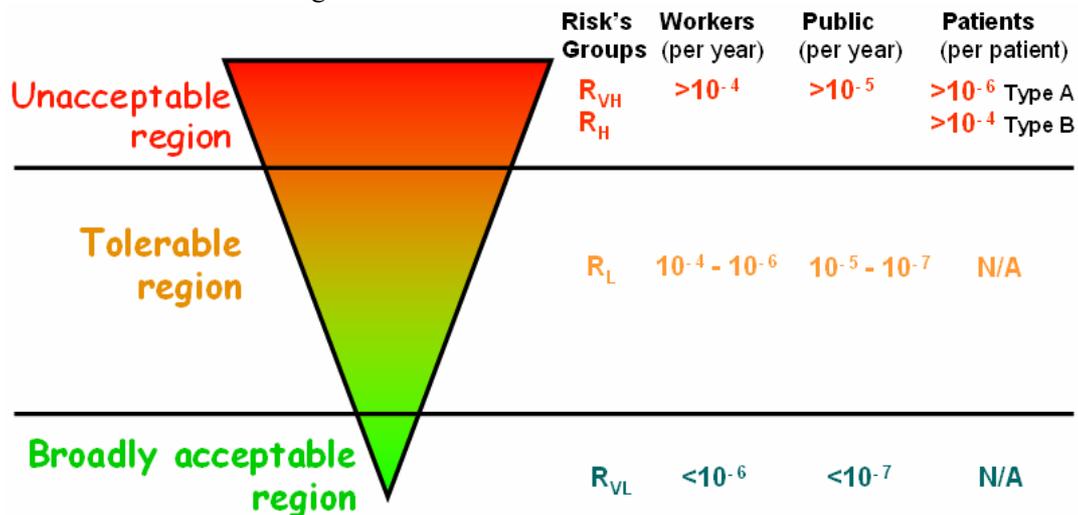
The PSA combines risk analysis tools to perform a systematic, exhaustive and structured investigation of the various scenarios, which may lead to undesirable consequences (accidental event sequences) starting from equipment faults and human errors. The PSA provides qualitative and quantitative information on the minimum cut sets of equipment faults and human error, which can cause an accidental sequence, the frequency of occurrence of the accidental exposure and faults and errors, which contribute most to the risk and a comparison of options to identify those with higher effectiveness in improving safety.

The risk matrix approach is a semi-quantitative technique, which, instead of quantifying absolute risks, uses a scale of risk levels or bands, such as “very high, high, low and very low”, which is used afterwards in the stages of decision making and implementation of safety measures.

2.2.3. Result analysis and decision making.

At this stage, quantified results are evaluated and measures to reduce risks are proposed. Criteria for risk acceptability are used based on national regulations or international standards or recommendations. Figure 1 contains acceptability criteria for risk management taken from the literature [8-9].

Figure 1: Criteria for risk management.



Nevertheless, establishing goals in terms of absolute values has the disadvantage of leading to promoting compliance with the absolute target values rather than focusing on the insights that the safety analysis may reveal. In addition, there are considerable uncertainties associated to the data and models used in the quantification, which render absolute value analysis less relevant than the relative contributions to risk. This facilitates identification of the elements that are more important to safety.

When PSA is applied, importance and sensitivity analysis reveal how many times a risk can be reduced or increase when a given defense is added or eliminated, which provides a powerful tool for decision making. When the risk matrix method is applied, result analysis shows how the risk band can change when some defenses are added or eliminated. In this case, sensitivity analysis allows evaluation on how the risk moves from one level to next when the level of one of the independent variables is changed.

2.2.4. *Implementación of risk-reducing measures*

At this stage, measures proposed in the previous step are applied. Since implementation of safety measures is to be done at a real facility, and the safety assessment of this study was done for a hypothetical facility, the last stage was not in the scope of this work. However, once the risk analysis has been performed, the high-level administration of the hospital would have a list of priorities to reduce the risk of the facility and be able to put the method in practice.

2.3. Application of risk quantification techniques to radiotherapy.

The Ibero American Forum of nuclear and radiation safety regulatory agencies (the FORO) has decided to adapt safety assessment techniques to hypothetical radiotherapy facilities, with the objective of promoting the application to the FORO's countries. In that sense, three applications have been carried out so far: 1) application of PSA to radiotherapy treatments with linear accelerators, 2) application of the "risk matrix method" to external beam therapy treatments with ^{60}Co and 3) to brachytherapy treatments.

2.3.1. *Probabilistic safety assessment methods.*

The probabilistic safety assessment combines different tools in a proactive research of the accidental sequences that can occur from a combination of equipment fault and human errors. PSA involves three fundamental tasks: identification of initiating events that may trigger accident sequences, determination of these sequences and calculation of frequency of occurrence.

The identification of initiating events was performed using Failure Modes and Effects Analysis (FMEA), a standard method to identify potential failure of an equipment, system or process and to analyze the resulting effects. Once the initiating events have been identified, the sequence of events that can evolve if no obstacle stops this development, i.e., if safety measures fail to work. To track and visualize these sequences "event trees" are drawn.

Once the event sequences have been drawn, the frequency of occurrence of the accidental exposure from each sequence is quantified. This frequency is computed by multiplying the frequency of the initiating event by the probability of failure of all safety measures involved in the sequence. The most significant events were determined through sensitivity and importance analyses, i.e. evaluation of how many times a risk is reduced or increased when a safety measure is added or removed.

2.3.2. *Risk matrix methods: the concept and the risk matrix table*

Similar to PSA, the risk matrix approach begins with identifying possible equipment faults and human errors that potentially lead to an accidental exposure. For this purpose, the tasks performed in each step of the radiotherapy process are analyzed and equipment faults and human errors in each of the tasks are identified.

The risk matrix approach is kept much simpler, as compared with PSA, in order to be usable in each individual radiotherapy department. The quantitative assessment of probabilities is replaced by a simpler, semi quantitative, four-level scale (for example, very low, low, high and very high) and the complex algebraic analysis of the event sequence done in PSA is replaced by a logical combination of the four levels of the frequency of the initiating event, the likelihood of failure of the safety provisions and the severity of the consequences. This combination results in a global risk for each initiating event. The global risk is also four-level scaled. The logical combination can be understood by means of the following three combination examples.

- For a potential error (initiating event), which is very unlikely to occur (frequency very low), and its consequences are very low, the risk may be negligible. There is no much to worry about an event sequence with low effects, which, in addition, are very unlikely to be caused. The result of the combination is "very low risk".
- The opposite can be said of an event sequence of very severe consequences and high likelihood of occurrence. This event sequence would raise much concern, the risk should be considered to be "very high", as there is a need to add safety measures to drastically reduce the risk by making the accidental exposure extremely unlikely.

- Not so obvious is the case of an event sequence, which is very unlikely to occur, but the consequences are very severe. Should the risk be considered “high”? Should preventive measures be applied? The answer is, yes. An event sequence with very severe consequences can not be left unattended, even if it is very unlike to occur, i.e., the frequency is very low. In this case, the risk is also considered “high”, and it is necessary to ensure that the event sequence be stopped by means of safety measures in order to prevent the very severe consequences.

This type of logical thinking is applied to all possible combinations of likelihood and severity and the results expressed in tabular form, called “risk matrix” (table 1). In the table, f stands for frequency, C for consequences, P for probability of failure of existing safety measures, and R for risk. The subscripts refer to the scale, VL stands for very low, L for low, M for medium, H for high and VH for very high. With regards to the consequences, VS stands for very severe; S for severe; MO for moderate and “mn” for minor. The resulting table is presented below:

Table 1: Complete risk matrix containing all combinations of the four levels of frequency of occurrence of the initiating even (f), the four levels of probability of failure of the set of safety measures (p) and the four levels of severity of consequences (C), if the initiating event results in an accidental exposure. The last column provides the four-level risks.

f _H	p _H	C _{VS}	R _{VH}	f _H	p _H	C _S	R _{VH}	f _H	p _H	C _{MO}	R _H	f _H	p _H	C _{mn}	R _L
f _M	p _H	C _{VS}	R _{VH}	f _M	p _H	C _S	R _H	f _M	p _H	C _{MO}	R _H	f _M	p _H	C _{mn}	R _L
f _L	p _H	C _{VS}	R _H	f _L	p _H	C _S	R _H	f _L	p _H	C _{MO}	R _L	f _L	p _H	C _{mn}	R _L
f _{VL}	p _H	C _{VS}	R _H	f _{VL}	p _H	C _S	R _H	f _{VL}	p _H	C _{MO}	R _L	f _{VL}	p _H	C _{mn}	R _L
f _H	p _M	C _{VS}	R _{VH}	f _H	p _M	C _S	R _H	f _H	p _M	C _{MO}	R _H	f _H	p _M	C _{mn}	R _L
f _M	p _M	C _{VS}	R _H	f _M	p _M	C _S	R _H	f _M	p _M	C _{MO}	R _L	f _M	p _M	C _{mn}	R _L
f _L	p _M	C _{VS}	R _H	f _L	p _M	C _S	R _H	f _L	p _M	C _{MO}	R _L	f _L	p _M	C _{mn}	R _{VL}
f _{VL}	p _M	C _{VS}	R _H	f _{VL}	p _M	C _S	R _L	f _{VL}	p _M	C _{MO}	R _L	f _{VL}	p _M	C _{mn}	R _{VL}
f _H	p _L	C _{VS}	R _H	f _H	p _L	C _S	R _H	f _H	p _L	C _{MO}	R _L	f _H	p _L	C _{mn}	R _{VL}
f _M	p _L	C _{VS}	R _H	f _M	p _L	C _S	R _H	f _M	p _L	C _{MO}	R _L	f _M	p _L	C _{mn}	R ^{VL}
f _L	p _L	C _{VS}	R _L	f _L	p _L	C _S	R _L	f _L	p _L	C _{MO}	R _L	f _L	p _L	C _{mn}	R _{VL}
f _{VL}	p _L	C _{VS}	R _L	f _{VL}	p _L	C _S	R _L	f _{VL}	p _L	C _{MO}	R _L	f _{VL}	p _L	C _{mn}	R _{VL}
f _H	p _{VL}	C _{VS}	R _H	f _H	p _{VL}	C _S	R _L	f _H	p _{VL}	C _{MO}	R _L	f _H	p _{VL}	C _{mn}	R _{VL}
f _M	p _{VL}	C _{VS}	R _L	f _M	p _{VL}	C _S	R _L	f _M	p _{VL}	C _{MO}	R _L	f _M	p _{VL}	C _{mn}	R _{VL}
f _L	p _{VL}	C _{VS}	R _L	f _L	p _{VL}	C _S	R _{VL}	f _L	p _{VL}	C _{MO}	R _{VL}	f _L	p _{VL}	C _{mn}	R _{VL}
f _{VL}	p _{VL}	C _{VS}	R _L	f _{VL}	p _{VL}	C _S	R _{VL}	f _{VL}	p _{VL}	C _{MO}	R _{VL}	f _{VL}	p _{VL}	C _{mn}	R _{VL}

2.3.3. Strengths and limitations of the risk-analysis techniques used in this study

Probabilistic Safety Assessment.

PSA provides quantitative information with numerical values about how much a given safety measure reduces the risk or how much the absence of a given safety measure may increase the risk. With this information, the cost-benefit analysis is objective and precise. PSA identifies common cause to more than one event sequence, and points out the importance and priority of the safety measures to treat the common cause. Moreover, PSA combines several tools to evaluate safety (qualitative, quantitative, and graphical) which allow complementary inputs to cover the limitations of each tool, if used separately. In spite of the fact that PSA is an ideal technique for safety assessment, its application is

very complex, demands a time-consuming work and requires experts from outside the hospital. This has been the reason why PSA that very few studies have been performed for this type of facilities.

Risk matrix assessment.

The method is relatively easy to apply by any individual radiotherapy department. Once the initiating events and typical safety measures are identified for a generic radiotherapy department, any individual hospital can perform a tailored self evaluation. Although the risk matrix approach does not provide numerical values, it classifies events in four risk levels or bands, which facilitates screening for importance and helps allocation of resources and priorities.

Both methods are complementary

All methods have in common the task of identifying initiating events and typical safety measures. Effort invested in one of them can be to a large extent be used for the other, although it is not strictly necessary to perform both. For instance, initiating events and safety measures identified by FMEA for a PSA study can be used for any other method and vice versa, and in particular for the risk matrix assessment.

3. Conclusions

Up to present, safety assessment has been performed by incorporating lessons from reported accidental exposures (reactive approach). However, accidental exposures continue to occur in the practice of radiotherapy, thus indicating that other equipment faults and human error than those reported are possible. These faults and errors can be anticipated by a proactive systematic approach.

Risk analysis techniques are a valuable resource for risk identification, quantification and risk reduction management. The Ibero-American Forum of Nuclear and Radiation Safety Agencies has promoted the adaptation and use of these techniques to radiotherapy treatments with accelerators, ⁶⁰Co and brachytherapy.

In this work, a way to introduce risk analysis techniques to radiotherapy has been developed, to supplement prescriptive and reactive approaches. Two proactive methods, traditionally used in conventional industries, the PSA and risk matrix, have been adapted for radiotherapy treatments.

Risk analysis tools contribute to identify vulnerable aspects of radiotherapy treatments and provide a fundament for decision making in choosing safety measures, with the help of quantitative criteria showing relative impact of these defenses on risk reduction. They are also useful to improve quality assurance in medical exposure for both hospitals and regulatory bodies.

The positive statements made on proactive methods should not be construed to conclude that one method is to replace the others, but rather the strength of these approaches resides in the synergy among them to improve overall safety in radiotherapy practice.

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Events with Radiological Significance in the Nuclear Industry. Unforeseen Personnel Exposure

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EVENTS WITH RADIOLOGICAL SIGNIFICANCE IN THE NUCLEAR INDUSTRY

UNFORESEEN PERSONNEL EXPOSURE

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1- Abstract

On September 1, 2005, Atucha 1 NPP, a 357 MWe PHWR was operating at a power level of 90% full power. While carrying out maintenance work related to replacement of fueling machine in use, for the one in stand by, a heavy water spill took place. As a result of it, some agents were contaminated. The evaluations and measurements of the received doses determined that only one agent of them received a significant dose. In fact, on Monday September 5, a communication of significant event was received at Nuclear Regulatory Authority from Atucha 1 NPP, notifying that the agent had received, according with preliminary information, an effective dose due to an internal incorporation of tritium that exceeded, significantly, the annual allowed limit. At the moment the evaluation continue to determine the root causes of the event. With the addition of the dose received by the agent on September 1, he overcame the dose limits constraint for the present year as well as the dose limits over the last 5 years

2- Description

The event happened during the execution of maintenance tasks in the refueling machine (RM); a spill of heavy water took place and as a result of it, some agents were exposed to a non planned intake of tritium. One of the agents exceeded the annual dose limit as well as the dose limits authorized by regulatory standards over the last 5 years.

Three Root causes and four Contributing Causes were identified. The method applied to conduct the evaluation, connected the causal factors for the different identified causes, with the levels of responsibility of the organization. It was observed in the mentioned scheme of evaluation that diverse causal factors exceeded the barriers that must have acted in each one of the previous levels

The utility, through note NASA-232/05, communicated Nuclear Regulatory Authority, the decision to carry out changes in the maximum levels of conduction of Atucha-I. The justification of this decision was based on “a progressive deviation of the practices that must be strictly respected to maintain the quality required operating and performing maintenance tasks into a nuclear power plant”. The performed changes were consistent with the following areas: Management, Operation, Radiation protection, Maintenance and Production.

Event description

Stage 1: Preparation tasks

On August 22nd preliminary tasks to replace the RM in operation for that one in stand by had begun. On August 31st the second RM was put in operation and soon after some abnormalities appeared which determined the need to make several calibrations in it. To achieve this, the Operation Department released a series of Warning Failure (WFs) These maintenance jobs required to put the RM out of service, by setting it to “test”

According to analyzed information, deficiencies in communication of tasks between 2 operation shifts and shift mates were detected. These lead to an inadequate preparation of the RM for the tasks to be developed during next day

Stage 2: RM tasks during the event day

During the briefing at 07:30 in the morning of 01/09/05 the maintenance tasks to be done during the day were discussed.

type	Number	Sector	Equipment	QC	Detail
WF	121478	ME	PL01M011	Y	Calibrate drum / flask position
WF	121479	ME	PL01M008	Y	Calibrate action disk
WF	121465	ME	PL01S022	N	Review open end of stroke
WF	121469	I&C	PL01M005	N	Review calibration
WF	121470	I&C	PL01M025	N	Review calibration
WF	121464	I&C	PL11P001	N	Review calibration
WF	121474	I&C	PL01P001	N	Review calibration

Table 1: List of tasks authorized in the RM by the Shift Head

After initiated the authorized tasks, the Refueling System Operator (RSO) considered that it was a good chance to verify the calibration of the air position indicator of the lock of the drum of the RM (component identified as PL01M003). This task was not authorized, therefore did not have the

corresponding WF and Work Authorization (WA) signed, nevertheless the RSO decided (jointly with the Supervisor of Instrumentation and Control) to do it.

Therefore, during the morning several tasks of electrical maintenance and instrumentation and control were made in the RM, like Calibration of drum / flask position (task 1), simultaneously with the “Review open end of stroke of valve PL01S022” (task 2)

Task 1 execution began to be achieved by personnel from electrical maintenance and instrumentation and control requested for assistance from Mechanical Maintenance Supervisor (MMS) and 2 more technicians who were working in other area of Controlled Area. To carry out task 1 they opened a hood (hood corresponding to the PL01M003) and placed a hose with extension to connect the RM to a system to produce depression in the RM (TY system). After finished task 2, Radiation Protection Officer (RPO) (in Control Room) opened the valve, generating pressurization of the RM with the out coming of gas and heavy water blowing through PL01M003. As a consequence several people were spread. An immediate personnel evacuation took place, without finishing task 1 including the hood that was out of its place.

Note: During the evacuation the corresponding acoustic alarm was not emitted. This alarm is essential to prevent the potential access of non-authorized personnel into the spill place.

Stage 3: Tasks of Cleaning, Drying and Spill Control

After the spill of heavy water, a RPO acceded to the area in order to clean-up and dry the spilled heavy water; he was accompanied by the MMS. In addition 2 other agents entered to try to stop the spill by valves operation.

MMS observed the drying job near the entrance of the RM area. During the drying, the RPO lift up the hood, and maintained it with one hand whereas with the other one he continued with the drying. Observing this situation MMS suggested to place the hood in its original position; the RPO agreed, but without fitting it

When executing this partial closing, a change in the conditions of pressure occurred, and as a result of it, a continuous heavy water spill started through the hood. When detected this new spill, all the personnel nearby evacuated the RM area. At that time, the evacuation acoustic alarm was not emitted either

Due to tritium presence in RM area, the intervening maintenance personnel in the second stage of the event left the place. On behalf of this situation, the MMS decided to enter the area to fit the hood, without a previous planning. Before entering the reactor building, the RPO dressed the MMS with a protection suit, with independent breathing equipment.

MMS got into the area alone and immediately he listened the alarm that indicated “air tube empty”. In this condition, he decided to put out the independent equipment of breathing and wearing no mask he fit the hood. After that he left the reactor building.

Task 4: Recovery Actions

At this point, RPO suspected that the heavy water came from the system TA, consequently asked Operations to close the valve identified as TW10S01 that meant that the leak was coming through the hose unions. The heavy water spill finally was controlled when Mechanical maintenance personnel entered the RM area to close valve PL01S090 and retired the mentioned hose

Later, the evaluations and measurements of received doses determined that only one agent received significant dose. This person was MMS, who received an effective dose by internal tritium intake of 41.8 mSv. The mentioned agent, who had at July 2005 an accumulated 12.63 mSv annual dose and 92.13 mSv in five years, exceeded the annual and five years limits authorized by Regulations

3- Assessment

Root and Contributing Causes of the event

The event evaluation showed three Root and four Contributing Causes

Afterwards the deficiencies and violations detected were mentioned and linked with Root and Contributing causes identified

RC 1

Failure in programming, coordination and execution of maintenance and calibration tasks: Lack of procedures. WF without properly processed WO and un identifying Orders to operate (OPO' s), Lack of work plans and Instructions

RC 2

Failure in preparation of a set of instructions to act in a coordinated way in situations which involve heavy water spills without having previously identified the situation of the related systems and the possible causes that originated it

RC 3

Failure in barriers represented by calibration and maintenance procedures and/or its properly use. Q.A allowed to perform tests and/or calibration without the required authorization represented by procedures

CC 1

Failure in the implementation of adequate measures to guarantee the fulfillment of Regulatory Requirement N 12 (related with dose constraints).

CC2

Failure in maintenance and renewal of obsolete radiation protection equipment

CC3

Deficiencies in transmission of pending tasks between operation shifts, deficiencies in definition of priorities and preparation of instructions about the condition of the RM before the work

CC4

Weaknesses in the training of the correct use of the equipment and techniques of radiation protection, and in the selection of the suitable protection clothes

Causal Factors:

1- Related with RC N° 1:

-Violation of the established in CNAI Procedure PI-16 "Management of tasks in CNA-I", with respect to the aggregate of unplanned and unauthorized task of maintenance. This deviation was identified as one of the causal factors of the root cause CR1.

2- Related with RC N° 2:

No application of a contingency plan that contemplates measures of suitable protection to control the loss of heavy water, acting without previous identification of the real situation of the systems and the possible causes of the leak. Actions were taken prioritizing the control of the spill of heavy water over the personnel safety. According to the analysis of identification of the root and contributing causes of the event, this deficiency will be identified as the root cause RC2

3- Related with RC N° 3:

Weakness in later controls of maintenance and calibration processes, that allowed the accumulation of Warning Failures (WF's) pending in the RM which recently had been qualified. According to the analysis of identification of root

and contributors causes of the event, this deviation was identified as one of the causal factors of the root cause CR3.

4- Related with CC N° 1:

Non fulfillment of the Regulatory Requirement RQ-NASA-012. According to the evaluation, at least two of the agents involved in the development of the event were under the control required by the RQ-NASA-012, but in both cases the effective application of the demanded control was not verified. According to the analysis of identification of the root causes and contributors of the event, this deviation was identified like contributing cause CC1

5- Related with CC N° 2:

Failure in maintenance and renewal of obsolete radiation protection equipment. According to analyzed information, one of the causes of the limited operation of the independent equipment of breathing was that the tubes of the equipment were not loaded to the maximum of their capacity, because due to their antiquity the required pressure test were not made. According to the analysis of identification of the root causes and contributors of the event, this deficiency was identified like the contributing cause CC2

6- Related with CC N° 3:

Deficiencies in the communication of pending tasks of maintenance between operation shifts and members of the crew. According to the analyzed information, the conditions were not transmitted with the due clarity in which the RM had been prepared. This lead to an inadequate preparation of the systems for the accomplishment of the tasks to be developed. According to the analysis of identification of the root causes and contributors of the event, this deficiency was identified like contributing cause CC3.

7- Related with CC N° 4:

Deficiencies in the use of radiation protection equipment and techniques. The time of use of the air tube depends, among other factors, of the appropriate use of the regulating valve. According to the analyzed information, there existed weakness in the qualification and training of personnel in the use of protective equipment. The investigation made allowed to extend this conclusion to other agents who potentially can require their use, in addition to the irradiated one. According to the analysis of identification of the root causes and contributors of the event, this deficiency was identified like contributing cause CC4

Root Cause (RC)	DESCRIPTION
RC1	Failure in programming, coordination and execution of maintenance and calibration tasks: Lack of procedures. WF without WO properly processed and identifying Orders to operate (OPO' s), Lack of work plans and Instructions
RC2	Failure in the preparation of a set of instructions to act in a coordinated way in situations which involve heavy water spills and without to have previously identified the real situation of the related systems and the possible causes that originated it
RC3	Failure in barriers represented by calibration and maintenance procedures and/or its use. Q.C allows to perform tests and/or calibration without the required authorization represented by procedures
Contributing Causes (CC)	DESCRIPTION
CC1	Failure in the implementation of adequate measures to guarantee the fulfillment of Regulatory Requirement N 12 (related with dose constraints).
CC2	Failure in maintenance of radiation protection equipment and renewal of obsolete equipment
CC3	Deficiencies in transmission of pending tasks between operation shifts, definition of priorities and instructions about the condition of the RM before the work.
CC4	Weaknesses in training related with the right use of radiation protection clothes and equipments
Direct Cause (DC)	DESCRIPTION
DC	Continuous loss of heavy water with tritium through PL01M003 partially closed

4- Event Precursors / Recurrences

a)16-03-01. During Programmed Outage an agent received a dose superior to the annual limit allowed (1)

b) Internal event N° 12/04. The direct cause of this event was violation of internal procedures, because the tasks were executed without WO. The identified root causes were:

Unsafe work practices, Unsuitable habits developed by the pressure/culture of the group and "Taking allowed short cuts/tolerated". (2)

c) Internal event N° 04/05: Two of the event causes were identified as "lack of coordination to do the work (without WO)" and inadequate preliminary evaluation of exposure rate" (3)

d) Internal event N° 05/05: The W.O was emitted but dose received by personnel exceeded the allowed for the job. It was caused by changes made by supervision who increased the amount of tasks previously authorized. Root causes: "Inadequate preliminary evaluation of exposure rate" and "Inadequate supervision of work time" (4)

- (1) Internal event, external irradiation. Level 2. Date: 16-03-01
- (2) Tritium Intake, Internal event. Level 2. Date: 04-05-04
- (3) Radioactive Contamination. Level 3. Date: 21-02-05
- (4) Dose exceed the stipulated in the work order. Level 2. Date: 26-02-05

5 - Corrective action / lessons learned

The 11 corrective actions recommended by the Internal Safety Committee (CIAS) and the Committee of Technical Revision (CRT) were as follows:

- 1- Dissemination of the event to all the CNA-I sectors, enforcing the supervision of application of procedures PI 01, PI 16 and PS 16.
- 2- To prepare re training of operators of RM, shift heads and assistants of shift heads in "procedure to operate number 5" (POP05)
- 3- To prepare retraining for radiation protection personnel in accomplishment of plant documentation requirements, specially in corrective actions and Manual of Missions and Functions
- 4- To prepare retraining for maintenance personnel in application of procedures PI 01 (Order to operate), PI 16 (Management of tasks in CNA-I) and PS 16.
- 5- To prepare retraining for Q.A personnel in application of procedures PI 01 (Order to operate), PI 16 (Management of tasks in CNA-I) and PS 16 (Work Authorization).
- 6- Manager of CNA-I would define the policy to follow for allocation of Administrative Resources
- 7- To prepare a retraining program for supervisors and technicians in the use of radiation protection equipments
- 8- Manager of plant will introduce this event in the opening of the next courses of safety.
- 9- Operation Section will carry out a pilot experience in self-evaluation
- 10- To define actions to be implemented with personnel that have dose limitations. Establish limits of 90 mSv in 5 years.

11- To analyze the possible modification of pipe TY 11 (make a prolongation) towards the RM maintenance room

More Corrective Actions included specific retraining for agents involved in the event.

CNA-I Management introduced other changes in personnel and procedures of Production Sector, incorporating agents with experience in the operational areas of the primary and secondary systems. In addition, in the daily briefings the participation of Supervisors was replaced to avoid situations that occurred in the past, in which the Supervisor which attended the meeting was different to the one that participate in the task.

CNA-I updated procedure PP-08 "Programming of Tasks" considering WANO recommendations, whose objectives are to define and to establish the methodology for the coordination, programming, liberation, follow up and control of all the tasks in the CNA-I, in any operative condition of the facility. This procedure introduced changes in the daily and weekly programmed tasks.

Additionally CNA-I updated procedure PP-09 "Management of Tasks in CNA-I", transferring procedure PI 16 from Engineering to Production, with the intention of guarantee the adequate closing of daily / weekly programming before the execution of tasks.

Afterward, some changes were made in the management staff of the facility.

5.2- Measures taken by Regulatory Authority CNA-I / Recommendations

In the report of evaluation of the event prepared by the Group Ad-Hoc created by ARN it was indicated that the corrective actions taken by the plant were applicable to avoid recurrence of Identified Causes.

ARN accepted recommendations made by group Ad Hoc related with regulatory actions

5.2.1 Follow up of implemented corrective actions through Inspections, evaluations and audits

5.2.2 Analyze the convenience of requiring the application of a procedure with instructions to act in coordinated way in the handling of operative incidents with associated radiological risk.

5.2.3 Analyze the convenience of the positions of Production Head and Quality assurance Head would be specified functions, considering the importance of the

tasks associated to the sectors Production and Quality assurance and its relation with the event.

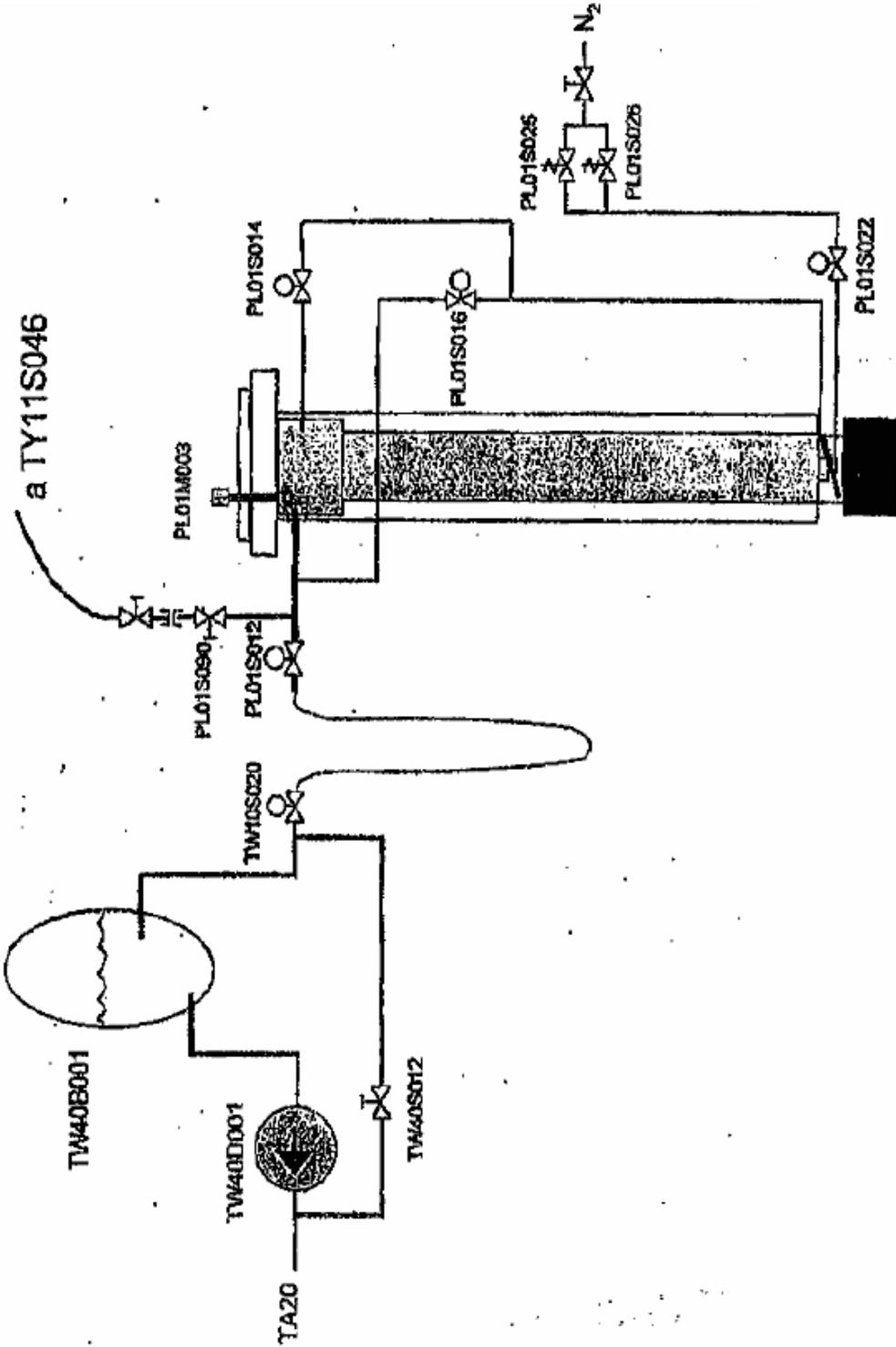
5.2.4 Require to CNA-I the complete revision of Work Plans and Orders to Operate related to the maintenance and calibrations of the RM, considering that during the investigation, weaknesses in the content of Work Plans and Orders to Operate were detected (and in some cases lack of them).

5.2.5 Follow up the project “Modification to the Installation” that replaces the used hose to generate depression in the RM, by a fixed pipes system.

6- REFERENCE

ARN: RN-IT-10/06: Evaluación Técnica Final del evento interno N° 21/05 “Exposición no planificada de personal por incorporación de tritio” en la CNA-I

ATTACHMENT: SIMPLIFIED SCHEME OF REFUELING MACHINE SYSTEMS



Application of IAEA Criteria for Internal Monitoring of Occupationally Exposed Workers of Nuclear Medicine

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Application of IAEA criteria for internal monitoring of occupationally exposed workers of nuclear medicine

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Abstract. According to the IAEA's BSS 115 individual monitoring should be undertaken, where appropriate adequate and feasible, to any worker who normally works in a controlled area and may receive a significant exposure. Such qualitative criteria should be applied both for internal and external exposures as the handling of unsealed sources usually involves both kinds of exposures. The radiation protection plan submitted by the Installation to the Regulatory Board of each country as part of the licensing process shall to include an evaluation of both kinds of exposures, together with a description of the measures taken to control them (e.g., shielding, containments, ventilation) as well as the measures aimed at keeping the exposures as low as possible and at implementing the planned individual monitoring program. The IAEA suggests, in its publication RS-G-1.2, the quantitative criteria for the evaluation of the need for the implementation of a program for internal monitoring; (i.e. that internal monitoring should be carried out whenever the potential internal exposure of incorporation leads to a value of annual committed effective dose equal or higher than 1 mSv) and the intake estimating methodology. This work presents a simulation of the application of such methodology for iodine 131, taking into account standard handling conditions in nuclear medicine services. Based on the IAEA methodology, the activity of iodine 131 has been calculated from which internal monitoring should be implemented in a routine basis. It can be concluded that, for internal monitoring the proposed methodology may be excessively restrictive in most cases when applied to nuclear medicine practices. The introduction of additional correction factors in the general formula of the methodology is suggested in order to include the following parameters: (1) fraction of handled activity by each worker; (2) the individual workload and (3) an intake factor of 10^{-4} . Furthermore, the calculation of the decision factor should be individualized for each worker, considering only the fraction of the total handled activity by each of them. These modifications would make the methodology more realistic and would permit its application as a basis for planning the implementation of internal monitoring programs in nuclear medicine services.

KEYWORDS: *Internal dosimetry, nuclear medicine, individual monitoring*

1. Introduction

In assessing whether individual monitoring of workers in an installation is required, the magnitude of potential internal exposure that could result during the routine process has to be determined.

The results from previous monitoring (individuals or workplace), when available from similar facilities, are of great value to make decisions and they are relevant for an "a priori" analysis to estimate the magnitude of internal exposure. The determination of radiological parameters in a facility is essential to assess the real situation and to identify those items which need an effort to improve for the radiological protection of workers and to minimize the potential internal exposure. The decision of implementing individual monitoring based on the results of previous measurements is the most valuable procedure for the most appropriated design.

According to the IAEA's BSS 115 [1] individual monitoring should be undertaken, where appropriate adequate and feasible, to any worker who normally works in a controlled area and may receive a significant exposure. Such qualitative criteria should be applied both for internal and external exposure

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and the unsealed sources handling usually involves both kinds of exposures. The radiation protection plan submitted by the Installation to the Regulatory Board as part of the licensing process shall to include an evaluation of both kinds of exposures together with a description of the measures taken to control them (e.g., shielding, containments) as well as the measures aimed at keeping the exposures as low as possible and at implementing the planned monitoring program. The IAEA suggests, in its publication RS-G-1.2, [2] the quantitative criteria for the evaluation of the need for the implementation of a program for internal monitoring; (i.e. that internal monitoring should be carried out whenever the risk of incorporation leads to a value of annual committed effective dose equal or higher than 1 mSv) and a methodology for a rough estimation of the expected intake, as well as a "decision factor" "d_j" for considering the necessity of implementation of individual monitoring to workers.

This work analyses this methodology through an application example involving ¹³¹I selection based on the possibility of higher internal exposure in nuclear medicine services, considering standard handling conditions. A modified methodology is suggested applying scenario-specific correction factors in the general IAEA formula.

2. Methodology description

The example was analyzed first by IAEA methodology and second by the IAEA modified methodology for ¹³¹I, which is a radionuclide of common use in nuclear medicine, considering the standard handling conditions in nuclear medicine services. Based on both methodologies, the activity of radionuclide that, if handled throughout a year would justify the implementation of an internal monitoring in a routine basis, was calculated.

2.1 The IAEA quantitative criteria

According to this methodology and based on the possibility of a committed effective doses of 1 mSv or higher in one year, a number of factors should be taken into account, such as:

- (i) The physical form safety factor f_{fs} , based on the physical and chemical properties of the material being handled. In the majority of cases, f_{fs} should be given a value of 0.01. However, in some cases, where it can be justified, a value of 0.001 may be used;
- (ii) The handling safety factor f_{hs} , based on experience of the operation being performed and the form of the material; and
- (iii) The protection safety factor f_{ps} , based on the use of permanent laboratory protective equipment (e.g. glove box, fume hood).

Suggested values of f_{hs} and f_{ps} for general application are given in Tables 1 and 2 respectively, but careful attention should be given to the circumstances affecting individual cases. The form of the material used (e.g. volatile liquid, powder) may sometimes be taken into account both directly (i.e. through f_{fs}) and indirectly, through the relative efficiency of the protective measures that are being taken (i.e. through f_{hs} and/or f_{ps}). The following tables illustrates how these factors may be applied to determine whether individual monitoring is required.

Table 1: Handling Safety Factors

Process Handling safety factor	f_{hs}
Storage (stock solution)	0.01
Very simple wet operations	0.1
Normal chemical operations	1
Complex wet operations with risk of spills	10
Simple dry operations	10
Handling of volatile compounds	100
Dry and dusty operations	100

Table 2: Protection Safety Factors

Protection measure Protection safety factor	f_{ps}
Open bench operations	1
Fume hood	0.1
Glove box	0.01

A specific radionuclide ‘decision factor’ d_j for a specific practice can be defined as:

$$d_j = \frac{A_j e(g)_{j,inh} f_{fs} f_{hs} f_{ps}}{0.001}$$

where

A_j is the cumulative activity, in Bq, of radionuclide j present in the workplace over the course of the year,

$e(g)_{j,inh}$ is the dose coefficient (Sv/Bq) for inhalation of radionuclide j (from BSS Table II–III [1], with the AMAD normally taken to be 5 μm for the workplace), and

0.001 is a conversion factor from Sv to mSv.

If f_{fs} has the default value of 0.01, the above equation may be simplified to:

$$d_j = 10 A_j e(g)_{j,inh} f_{hs} f_{ps}$$

Assuming a $d_j = 1$ mSv, the methodology described above, is used for calculating the threshold activity of the radionuclide above which internal monitoring should be implemented in a routine basis and the results are shown in this work.

2.2. IAEA modified quantitative criteria

The relevance of describing an “a priori” scenario previous to the analysis it is assumed for this proposal. In this way, each scenario could involve more than one worker but all of them being involved in similar tasks and periods of time to perform such task, so that, we will refer to “ $d_{j_scenario}$ ”.

Thus, the specific radionuclide ‘decision factor’ $d_{j_scenario}$ for a specific practice can be calculated with the same formula described above though with some additional correction factors:

$$d_{j_scenario} = \frac{A_j e(g)_{j,inh} f_{fs} f_{hs} f_{ps}}{0.001} \times f_{workload} \times f_{handled_activity} \times f_{resuspension}$$

The three proposed additional correction factors are the following:

$f_{workload}$: fraction of time involved in a particular task by the worker in the scenario.

$f_{handled_activity}$: the fraction of handled activity by the worker in a scenario considering that in real practice each worker, according to his responsibilities, could manipulate only a fraction of the total activity in the specific area.

f_{intake} : fraction of the handled activity that could be incorporated by the worker through aerolization or volatization.

The assignment of values to these parameters could be the following, according to the scenario analyzed:

f_{workload} : to be defined by the Radiation Safety Officer of the Installation according to the time assigned to the task, considering 2000 hs/year. The value to be assigned is ≤ 1

$f_{\text{handled activity}}$: fraction of handled activity in a given scenario

$$f_{ha} \cong \frac{\text{handled activity in a scenario}}{\text{total activity}}$$

f_{intake} : references [3], [4], [5], [6], suggest that, for the assessment of a potential internal exposure from the handled activity, it would be a good and still conservative approach to consider a fraction of 1×10^{-4} . It should be considered that the fluctuation of handled material values during the year has a range acceptable for the average value calculated to represent the annual handled activity

3. Example case analysis

3.1. Scenario description: ^{131}I Na for thyroid cancer

The scenario considers a single volatile compound ($f_{\text{hs}}=100$), ^{131}I Na, being handled [7] inside a fume hood ($f_{\text{ps}}=0.1$) and with the default value of $f_{\text{is}}=0.01$. ^{131}I Na was selected based on its broad use as an effective treatment for thyroid cancer and in its high amount of activity prescribed to each patient for ablation in the order of 3.7 GBq (100 mCi) to 7.4 GBq (200 mCi). It is a scenario with a possibility of significant internal exposure that could take place mostly during the fractioning of the therapeutic activity received weekly in a nuclear medicine service according to the prescription for each patient.

3.2. Scenario analysis applying IAEA criteria methodology

The activity that determines a value of $d_j = 1$ mSv is calculated by applying appropriate values of each parameter in the IAEA general formula as follows:

$$d_j = \frac{A_{131I} \cdot 2 \times 10^{-8} \cdot 0.01 \cdot 100 \cdot 0.1}{0.001}$$

$$d_j = A_{131I} \times 2 \times 10^{-8} \times 100$$

$$1 \text{ mSv} = A_{131I} \times 2 \times 10^{-8} \times 100$$

$$A_{131I} = 5 \times 10^5 \text{ Bq} = 0.5 \text{ MBq} (13.5 \mu\text{Ci})$$

This means that 0.5 MBq (13.5 μCi) is the handled activity in a year by a worker that would lead to an internal exposure above 1 mSv. It may be excessive, since a nuclear medicine service could manipulate iodine 131 in the order of 18.5 GBq (500 mCi) per week. Considering 50 week per year, the annual handled activity could be approximately be 925 GBq (25 Ci).

The available information concerning the assessment of internal exposure of workers in nuclear medicine departments [9] [10] [11] [12] y [13] shows that the intake of iodine 131 could be much lower than the prediction from IAEA's methodology. In this example, this formula considers that 10 % of the handled activity could be incorporated by the workers though this is an overestimation. Therefore a modified methodology is proposed including scenario-specific correction factors in the general formula that would make this methodology more realistic.

3.3. Scenario analysis applying IAEA criteria modified methodology

For the same scenario, the activity that determines a $d_j = 1$ mSv is calculated applying the proposed modified formula:

$$d_{j_scenario} = \frac{A_j e(g)_{j,inh} f_{fs} f_{hs} f_{ps}}{0.001} \times f_{workload} \times f_{handled_activity} \times f_{intake}$$

For $f_{workload}$ in this scenario the task is assumed to be performed once a week, 50 h/year, so that 50 divided by 2000 makes 0.025:

$$f_{workload} = 0.025$$

For $f_{handled_activity}$ it is assumed that the whole activity is handled, so that

$$f_{ha} \cong \frac{\text{handled activity in scenario}}{\text{total activity}} \quad f_{handled_activity} = 1$$

For f_{intake} it is assumed the conservative value mentioned above, so that, $f_{intake} = 1 \times 10^{-4}$

Replacing the values in the formula the results are:

$$d_{j_scenario} = \frac{A_{131I} \cdot 2 \times 10^{-8} \cdot 0.01 \cdot 100 \cdot 0.1}{0.001} \times 0.025 \times 1 \times 1 \times 10^{-4}$$

Assuming $d_j = 1$ mSv and re-arranging factors,

$$1 \text{ mSv} = A_{131I} \times 2 \times 10^{-8} \times 2.5 \times 10^{-4}$$

The handled activity in a year by the worker that could lead to an internal exposure above 1 mSv according to the proposed modified formula results in:

$$A_{131I} = 5 \times 10^{12} \text{ Bq} = 5 \text{ TBq} (135 \text{ Ci})$$

It might be assumed that workers responsible for fractionating therapeutic activities of ^{131}I inside fume hood in the order of 25 Ci per year are far from exceeding 1 mSv in the year. However, it should be considered that the criteria has been applied to a simplified scenario with the purpose of evaluating the methodology. Other information should complement the application of the criteria in real life.

4. Discussion

These quantitative criteria could be a useful tool to be applied by the Radiation Protection Safety Officers to analyze different scenarios in order to assess the influence of different radiation protection conditions to improve the handling processes to ensure workers' safety. This methodology could be helpful in the design of an individual monitoring plan to be submitted by the nuclear medicine service to the Regulatory Board as part of the licensing process since it should include an evaluation of internal exposure risks

This work presents an application example including only one radionuclide, but in real scenarios, the same worker may be assigned multiple duties involving the handling of other radionuclides. The decision whether such workers require individual monitoring should be based on a careful review of all workers' duties and mainly the different radiological conditions in each scenario. Detailed guidelines on workplace categorization and monitoring requirements are given in NRPB-M443 [8], which imply a much more complex analysis, whereas the IAEA quantitative criteria is a simplified approach.

As mentioned above, monitoring data (individuals or workplace) are of great value to confirm the decision about the need for the implementation of an internal monitoring program. The determination of radiological parameters in a nuclear medicine service is essential to know the real situation and to identify the items which need an effort to improve the radiological protection of workers and to minimize the potential internal exposures. The decision on implementing individual monitoring based on measurement results is the most valuable procedure for the appropriated monitoring program design.

In that sense, it is advisable to provide training to Radiation Protection Safety Officers on internal dosimetry (direct and indirect measurements and dose assessment), for instance in the frame of IAEA Regional Projects [14], and the possibility of extend it to implement protocols for the calibration of diagnostic devices to estimate workers intakes such as the gamma camera and the thyroid probe [15], [16]. In this way, the medical services would have an useful tool for a quick estimation of an intake if an accident takes place as well as the possibility to carry out routine individual monitoring of workers if necessary

5. Conclusion

This work presents the analysis of IAEA methodology for the evaluation of the need for the implementation of an internal monitoring program ; considering that it should be carried out whenever the potential internal exposure of incorporation leads to a value of annual committed effective dose equal or higher than 1 mSv.

The handled activity that would lead to a value of $d_j = 1$ mSv has been calculated, in a selected scenario based on the possibility of a significant internal exposure that could take place during the fractioning of therapeutic iodine 131 activities received weekly in a nuclear medicine service

The analyses of IAEA methodology through an application example of fractioning of ^{131}I , suggest that it may be excessively restrictive in most cases when applied to nuclear medicine practices.

The introduction of additional correction factors in the general formula has been proposed, which for the analyzed scenario, takes the following values:

- (1) fraction of handled activity by each worker, $f_{\text{handled activity}}=1$;
- (2) individual workload, $f_{\text{workload}}=0.025$; and
- (3) an intake factor, $f_{\text{intake}}=1 \times 10^{-4}$.

These results prove that an IAEA modified methodology applying scenario-specific correction factors in the general formula would make this methodology more realistic as a basis for planning the implementation of internal monitoring programs in nuclear medicine services.

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GIS Application in Atucha I Nuclear Power Plant Exercise Argentina, 2007

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GIS APPLICATION IN ATUCHA I NUCLEAR POWER PLANT EXERCISE ARGENTINA, 2007

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ABSTRACT

Geographic Information Systems (GIS) are tools applied to assist in the assessment and solution of many geographical related issues. Recently, their applications have been extended to the areas of disasters and environmental emergencies. GIS not only could be used as a diagnostic tool. Combined with adequate information and other tools capable to predict the transfer of pollutants in the environment and the associated impacts to the public, GIS could be used to support emergency planning and response.

The Nuclear Regulatory Authority (NRA) of Argentina has incorporated in 2003 the GIS technology like an innovative resource for its preparation and response activities in emergencies. For this, the NRA acquired the necessary technology (hardware and software) and the technical specialist who were joined to expert's team in the nuclear and radiological emergencies field. The GIS stays operative as support tool in the Emergencies Control Center of NRA.

In this paper, the use of GIS as a tool for analysis and advice is presented. The GIS is being used for preparation and development of nuclear emergencies trials and exercises, carried out on-site and off-site at the Nuclear Power Plant Atucha I Buenos Aires, Argentina, in cooperation with civil defense, national and state security and army forces and intensive public involvement.

The databases were conformed with information from different sources, including the result of interviews to different actors, as well as other local and national government agencies and forces. Also, educational institutions, local medical centers, etc., were consulted. The information was enriched with outings to field in the surroundings of nuclear power plant. The scope and the detail of the information for this exercise covers 30 kilometers surroundings the nuclear power plant, with a range of significantly different geographical and population conditions. When loading the information in the GIS, a classification scheme is applied and additional regional scale information relevant for the purposes of the GIS is provided..

The conclusion is that the simulations and predictions obtained with the GIS to define the scenarios whether to exercise the planned response actions agrees significantly with the actual scenarios observed. Due to this coincidence, the GIS has demonstrated that facilitates the appropriate planning of the exercise and it is also a good tool to carry out improvements during the step of planning and response, resulting in a optimal tool to aid the decisions making process in real time.

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INTRODUCTION AND OBJECTIVES

The Nuclear Regulatory Authority, NRA, under current legislation, Law 24,804, must approve the contingency plans for facilities, municipalities and provinces potentially involved. In this context, the RNA established an Emergency Intervention System in Nuclear NEIS, which is responsible for the preparation and response to a nuclear emergency and also coordinates and integrates all the civil organizations implicated in such situation.

The aim is to show the functionality of the Geographic Information System, GIS, in the Emergency Exercise of the Nuclear Power Plant Atucha I, as an important and efficient tool in the modeling of a real accident.

SCENARIOS

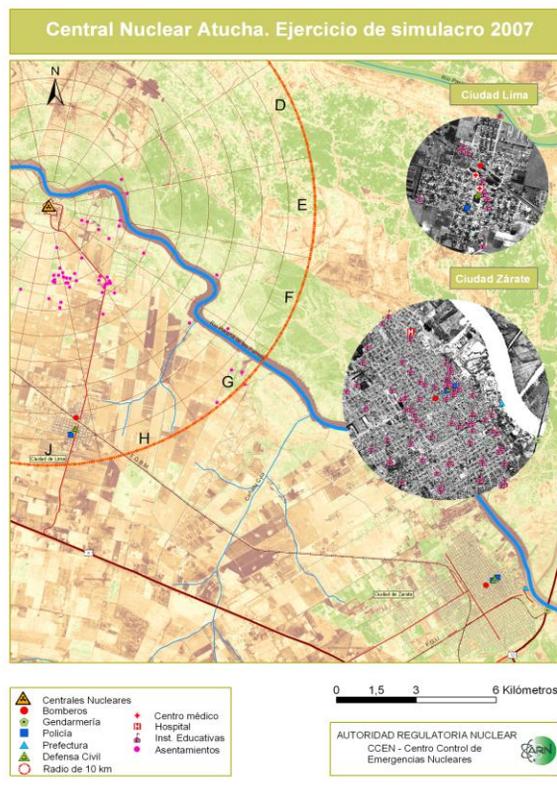
In each emergency exercise, is defined a number of scenarios that they will be executed; and the objective is to simulate a real emergency situation. The main stage is the creation of the Municipal Emergency Operations Center, EMCO. It conformed by the Civil Defense, Security Forces and Firefighter Brigades Volunteers, under the coordination of the NEIS. The role of EMCO is to implement automatic countermeasures in the eye of keyhole, these are: Making covered, Distribution of iodine tablets and Access Control, to establish the evacuation automatically within the 3km around the plant in all directions. These countermeasures are assessed as independent scenarios.



Access control



Distribution of iodine tablets



GIS FUNCTION

The main objective of the GIS developed in the RNA is to provide the necessary support for the response planning and the implementation of radiological protection actions during a nuclear emergency. The GIS provides the geographic data, the availability of logistics resources throughout the country. Incorporating the results of consequences models and environmental measurements into GIS, it allows the analysis of that information in real time and the display of results for decision making.

The geodatabase consists in: demographic, social and economic data, identification of key players implicated in the emergency response, road infrastructure, topography and natural features of the area in question, land use, mass media, schools, medical centers; etc. All necessary alphanumeric information (phones, addresses, personnel, vehicles, number of students, and so on.) as store in the database.

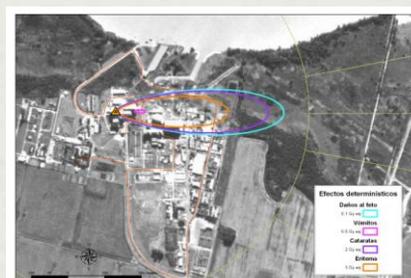
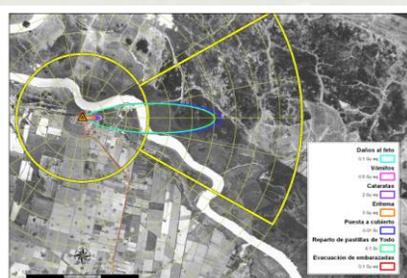
Another kind of information incorporated into of GIS are the satellite images and aerial photographs.

The GIS during the emergency assists in the management decisions, answering queries such as:

- Where the plume is moving towards and what are closest monitoring units
- What area is covered and how much population is affected
- Where should be applied the automatic countermeasures (sheltering, access control and evacuation)
- What water sources were potentially affected
- What is the rural area affected and the agricultural production involved

RESULTS ACHIEVED WITH GIS

The automatic countermeasures are taken in the keyhole. They coved have modifications depending on changes in the weather, the results of the consequences models and the field monitoring (using the information stored in the geodatabase).



¹The keyhole is formed from the prevailing wind direction, fulfilling an angle of 22.2° on either side of the vector direction and, in the first three kilometers in all directions. Acknowledgment: Civil Defense of Lima, Civil Defense of Zarate, National Gendarmerie, Police of Buenos Aires Province and Firefighter of Lima

The Need to Increase Regulatory Controls on Imports and Exports came from the Detection of an Argentine Import Containing Exceeding Pellets of Ir-192

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The need to increase regulatory controls on imports and exports came from the detection of an Argentine Import containing Exceeding Pellets of Ir-192

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Abstract. In Argentina, radioactive sources for application in industrial gamma radiography are manufactured using small metallic discs called pellets of radioactive isotope Ir-192, that are imported. They are to be shipped in specially sealed recipients according to their high radioactive levels. These recipients are large enough to contain activities as high as 296 TBq and are therefore approved as Type B(U) packages.

This paper thoroughly describes a particular import of this material in which an exceeding 220 ± 2 pellets (75,62TBq) was found. The total amount of activity had not been declared adequately by the consignor.

The corresponding notification to the Argentine Nuclear Regulatory Authority (NRA) was made by the importer and national manufacturer at the moment these particular pellets were to be used for source production. The exceeding material doubled that which had been authorized to be imported. After the notification and since the material had entered to the country illicitly, customs legislations were applied in a legal process in which the Nuclear Regulatory Authority participated as well.

The decision of the judge was to ship the exceeding cargo again to its proceeding country. However, a significant time had passed since the cargo arrived and the pellets had stuck to each other due to high temperature. This made them impossible to handle in order to encapsulate them as special form radioactive material again for their shipment in the available Type B (U) package.

In the outcome, the material was retained at the manufacturer facilities, in its original packaging, with suitable radiological and physical conditions to guarantee safety and security, waiting until its activity had decayed to acceptable levels for its shipment as Type A package.

As a result, a need to require further technical specifications from the vendor was detected. These specifications should warn about the period in which these pellets may be utilized for source production before sticking take place.

It is also necessary to make a feasibility study in order to establish a procedure that make it possible to verify from external radioactive measurement the activity consigned in shipments when importing and exporting. This would be of great help to perform the regulatory control of the incoming and outgoing radioactive material to and from any country.

KEYWORDS: *Import of Radioactive Material; Pellets of Ir-192; Shipment of Radioactive Material*

1. Introduction

1.1. The regulation of the Industrial Gamma Radiography activity in Argentina

The industrial gamma radiography is an area of work with a market consisting of some 60 companies around the Argentine Republic, which provide services to industries engaged in oil refining, pipeline construction and ship manufacturing, among other activities. These companies have different kinds of projectors of industrial gamma radiography. However, the suppliers of radioactive sources for these projectors are few.

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The Argentinean Nuclear Regulatory Authority authorizes in different ways the acquisition of radioactive material for the above mentioned use, which are listed below:

1. Through an authorization to import radioactive sources that is given to entities previously authorized to operate equipment of industrial gamma radiography for its own use.
2. It also authorizes companies whose purpose is to import and sale industrial gamma radiography sources.
3. Entities, previously authorized to operate equipment of industrial gamma radiography, may acquire industrial gamma radiography sources through a manufacturer, in Argentina, authorized to produce Radioactive Sources for use in industrial gamma radiography.

The amount of radioactive sources and radiation activity for which each company is authorized to handle is specified in terms of technical assessments related to radiation protection. These assessments are carried out by the technical staff, in particular inspectors, from the Argentinean Nuclear Regulatory Authority.

1.2. Radioactive Sources production

In Argentina, radioactive sources of the Ir-192 isotope are manufactured only to be used for industrial gamma radiography. For the manufacture, this isotope is imported in small metallic discs called pellets. These pellets are encapsulated as special form radioactive material and are shipped in Type B (U) packages. They are usually transported by air.

The manufacturer of radioactive sources has to ask for an authorization for the import of such material from the Argentine Nuclear Regulatory Authority. It must be declared the radioactive activity of the material at the estimated date of import. Once the radioactive material enters the country, the manufacturer, after the necessary arrangements with the customs, withdraws the container and moves it by land routes until its facilities.

At this point, the manufacturing process basically consists of the following steps:

The received container is placed inside a hot cell and the sealed capsules containing radioactive material are drawn.

The capsule is opened in order to manipulate the pellets, to build the sealed sources, through a technique developed by each manufacturer that will be used in the industrial gamma radiography equipments. These sealed sources are special radioactive form materials. Different radioactive sources are produced for the various designs of industrial gamma radiography projectors which exist in Argentina.

Finally, the empty container is returned to its proceeding country, to its owner.

2. Development

2.1. Pellets of Ir-192 detected as exceeding radioactive material in an Argentinean Import.

In March 2007 it was authorized, by the Argentinean Nuclear Regulatory Authority, the entry of 85.1 TBq of Ir-192 in pellets of 0.370 TBq each as it had been declared by the national importer and manufacturer.

As usual Argentina had given its prior consent to the proceeding country.[1]

The pellets came encapsulated in as special form radioactive material within a Type B (U) package. [2]

Later, the manufacturer notified to the Argentinean Nuclear Regulatory Authority that he had received a greater amount of radioactive material than the declared one. The amount was approximately twice of that declared. The exceeding material was 220 ± 2 pellets of Ir-192 with a total radioactive activity of 81.4 ± 0.740 TBq. Nevertheless, the type B (U) package design used, allowed a much greater radioactivity than the one that had been shipped. That fact allowed to be sure there had not been any radiological problem during the transport.

The Argentinean Nuclear Regulatory Authority performed an inspection in order to verify that the radioactive material was in adequate security and safety conditions. This inspection was carried on with Argentinean Customs. Customs were concerned with the fact it were a smuggling problem.

The exceeding pellets in the incoming container remain in the manufacturer facility under appropriate security and safety conditions.

Meanwhile this event caused legal actions from the part of Customs that involved the manufacturer. The manufacturer proposal of returning the exceeding radioactive material was the decision adopted. The Argentinean Nuclear Regulatory Authority, as well Customs agreed that was the best way to act. As a result of this, the manufacturer of pellets in the proceeding country, the Argentinean Nuclear Regulatory Authority and the national manufacturer of radioactive sources analyzed the ways in which this material might be sent back.

The main problem was that the original capsule containing the pellets had already been opened, and all pellets had been extracted in order to manufacture the radioactive sources. As a consequence, this capsule could not be used again to house the surplus pellets and then sealed it. Everything necessary for the normal relocation and transport as a Type B (U) package. [2]

It was then that the Argentine Nuclear Regulatory Authority in agreement with the national manufacturer decided to use these pellets and build radioactive sources according to the normal procedures and submit them as such to the manufacturer pellets in their home country. This would have been one feasible way to send the pellets in the special form required for the relocation in the Type B (U) package. [2]

When the national manufacturer wanted to use the Ir-192 pellets to manufacture radioactive sealed sources, it realized that these pellets were stuck to each other mainly by heat action. This made it impossible to handle them inside the hot cell to produce the radioactive sources.

This opened a new dilemma on what to do with this material to be able to forward it as soon as possible to their home country and, again, several options arose.

The first option was to send it as a special arrangement, which involved hiring an aircraft or other means of transport exclusively for the transport of this material, complying with the TS-R-1. It would have implied a big economical effort in getting the security and safety conditions required. This option was not taken into account, it was decided to explore another one.

The second option required that the manufacturer, from its home country, send an adequate capsule in which the pellets would be accommodated for the shipment in type B (U) package. This capsule should be sealed in Argentina.

However, the design of this capsule would have demanded an approval from the Argentinean Nuclear Regulatory Authority as an special form radioactive material [2]. This would have taken too long time, during which the radioactive material activity would have decayed by half.

This prompted the latest idea. On one hand, the radioactive material was in adequate security and safety conditions within the national manufacturer's facility. On the other hand, it was possible to wait until its activity decayed to a quantity that made it possible to make the transport in the same package that had entered to Argentina using it as a type A package [2]. Summing up, it would strongly simplify its transportation.

As a consequence, it was agreed that during its stay in the country, the material would be under the responsibility of the national manufacturer controlled by the Argentinean Nuclear Regulatory Authority under appropriate security and safety conditions. Then it was set the approximate date at which this material would have decayed to the radioactive activity that allowed the safe transport. Time in which it would be possible to return the exceeding material to its proceeding country.

2.2. A brief analysis of Incident

This incident did not have any radiological consequences. However, It would be desirable to strengthen the regulatory controls to detect easier undeclared incoming and outgoing of radioactive material.

In spite of good practices carried out by the manufacturer of sources in our country and its foreigner supplier of radioactive material, the control of documentation that has to be presented in order to import and/or export of radioactive material could be only a part of the control that should be done.

The Authors of the present work think that count on means to measure the radioactive activity with some precision, in order to verify the amount of it that has been declared in the corresponding exports and imports, it would be an improvement to the regulatory controls over radioactive material.

However, we know that for normal conditions of transport, other than special arrangements, the maximum dose rates in contact with package or overpacks should not exceed 2 mSv/h.[2]

Subsequently ¿can we measure with any degree of confidence dose rates lower than 2 mSv/h, and quantify them in order to use them in regulatory controls?

2.3. Quantifying the dose rate in contact with containers that use depleted uranium as shielding

Measuring the dose rate in contact with the external surfaces of packages or overpacks is a common task for operators and inspectors in order to categorize the packages, controlling this dose rate be less than 2mSv/h. [2]

Nowadays we propose to perform the same measurement to control the radioactive activity inside the package/overpack. We expect to make the same dose rate measurement, not only to provide with information to make the categorization or the control of the categorization, but to allow quantification of the radioactive activities. The dose rate measured will let to know, with a certain degree of precision and by a simple calculation, the range of radioactive activity found inside of the transport container. We will determine the calculation for a similar container and amounts of radioactive activity equivalents to that of the package that showed the control weaknesses and will see whether our method is feasible.

We will determine on a theoretical basis the dose rate in contact with a container that houses an radioactive activity of 85.1 TBq from Ir-192.

2.4. Relation between Activity and Dose Rate

The maximum dose rate measured in contact and outside of the container is directly proportional to the radioactive activity that carries inside. Consequently, the relation will be as follows:

$$D_{Ir-192}^{\bullet} = m \cdot A \quad (1)$$

Where

A = Activity of Ir-192 inside the container

D_{Ir-192}^{\bullet} = Maximum dose rate measured in contact outside the container in a given point

m = slope of the curve that only depends on the type of shielding that using the containers

2.5. Assumptions and Estimates

1.-The dose rate, from the container made of depleted Uranium shielding, is not considered by the calculation software but it is experimentally measured. When measuring the dose rate of the empty container, we obtained a value of 5µSv/h. It will be attributed to the presence of depleted Uranium in the shielding. To calculate the activity of Ir-192 inside the container, every experimental contact dose rate must be corrected using this criterion.

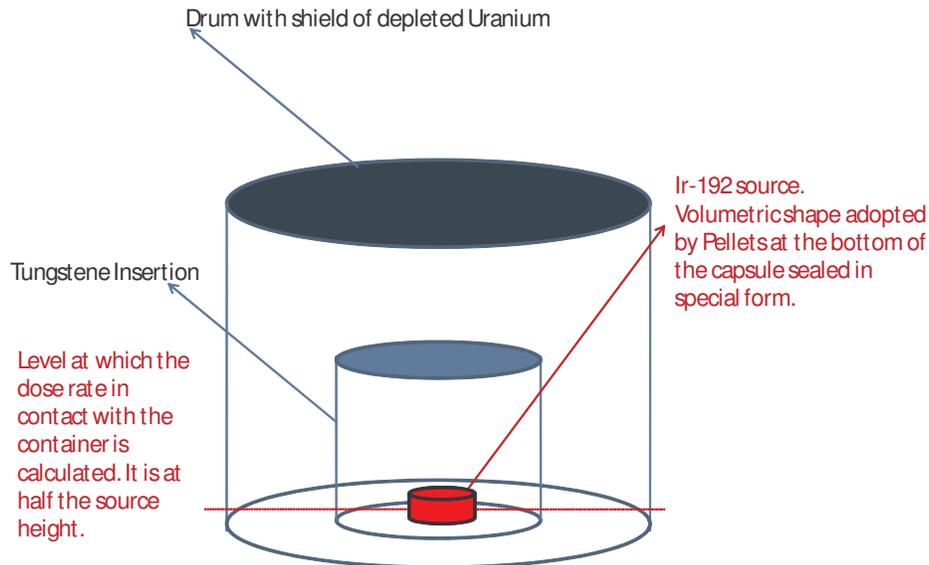
2.-We supposed that 85.1 TBq of Ir-192 are contained within one sealed capsule. The capsule is stuck to the tungsten insertion. The Ir-192 pellets are randomly located at the bottom of the capsule and they occupy a cylindrical volume which we consider as the geometrical form of the source.

3.- We subtract the experimental dose rate due to the depleted Uranium that is measured in the empty container.

4.-We adjust the activity range values considering the measuring error of the detector used.

2.6. Container Design

Figure 1: Transport Container Design simulated by software



2.7. Required data for the calculation

Table 1: Data taken on the basis of measures taken to the container, data taken from the Certificate of Approval of the type B (U) package and geometric approximations of the source made up of pellets of Ir-192

	Thickness (Cm)	Height (Cm)	Radius (Cm)
Source	-	0.33	0.45
Tungsten Insertion	2.05	-	-
Uranium shield	5.3	-	-
Uranium Cover	5.8	-	-
Activity of Ir-192 source	85.1 TBq in the 1st calculation		

MicroShield v.5.03 Software was used. Depleted Uranium was introduced as a factor of Build Up for the calculation in the program.

The dose rate spot was taken at X= 7.8 Cm, Y=0.165, Cm Z=0, where X represents the sum of all thicknesses and the source radius, and Y represents the height of the source.

2.8. Calculation Results

$$D_{total}^{\bullet} = D_{Ir-192}^{\bullet} + D_{U-235}^{\bullet} \quad (2)$$

From (2), we infer:

$$D_{Ir-192}^{\bullet} = D_{total}^{\bullet} - D_{U-235}^{\bullet} \quad (3)$$

Table 2: Experimental values and results obtained applying equation (3) and MicroShield Software

D_{total}^{\bullet}	$53 \pm 3 \mu\text{Sv/h}$ (a)
D_{U-235}^{\bullet}	$5 \pm 1 \mu\text{Sv/h}$ (b)
D_{Ir-192}^{\bullet}	$48 \pm 4 \mu\text{Sv/h}$ (c)
D_{Ir-192}^{\bullet} Theoretics	$44.90 \mu\text{Sv/h}$

- (a) Instrumental error attributed to Geiger - Müller detector fluctuation, used at the experimental measurement and whose fluctuation exceeded detection limit.
- (b) Instrumental error attributed only to detection limit of Geiger – Müller detector: $1 \mu\text{Sv/h}$
- (c) Sum of errors.

The software calculated theoretical value of D_{Ir-192}^{\bullet} falls within range values of experimental error, which confirms that the assumptions and estimation considered in 2.5 are valid.

The next step is to determine the slope m of equation (1) to be able to use the calculation in any range of activity. By doing so, we will be able to draw the characteristic curve for this particular container.

From (1) we infer:

$$m = \frac{D_{Theoretics}^{\bullet}}{A} \quad (4)$$

Using the theoretical value of D_{Ir-192}^{\bullet} obtained by software and the activity value of 85.1 TBq of Ir-192 in the container, we can infer the value of m at that dose rate.

Table 3: m value from equation (4).

m	$0.527 \mu\text{Sv}/(\text{h. TBq})$
-----	--------------------------------------

Considering this result for the slope, equation (1) becomes as follows:

$$D_{Theoretics}^{\bullet} = 0.527 \cdot A \quad (5)$$

When measuring activity in the container, we will measure a total dose rate as that indicated in table2:

D_{total}^{\bullet} . For this purpose, we need to consider the dose rate that comes from the container shielding:

D_{U-235}^{\bullet} , which has a constant value as indicated in table 2.

Equation (5) then becomes:

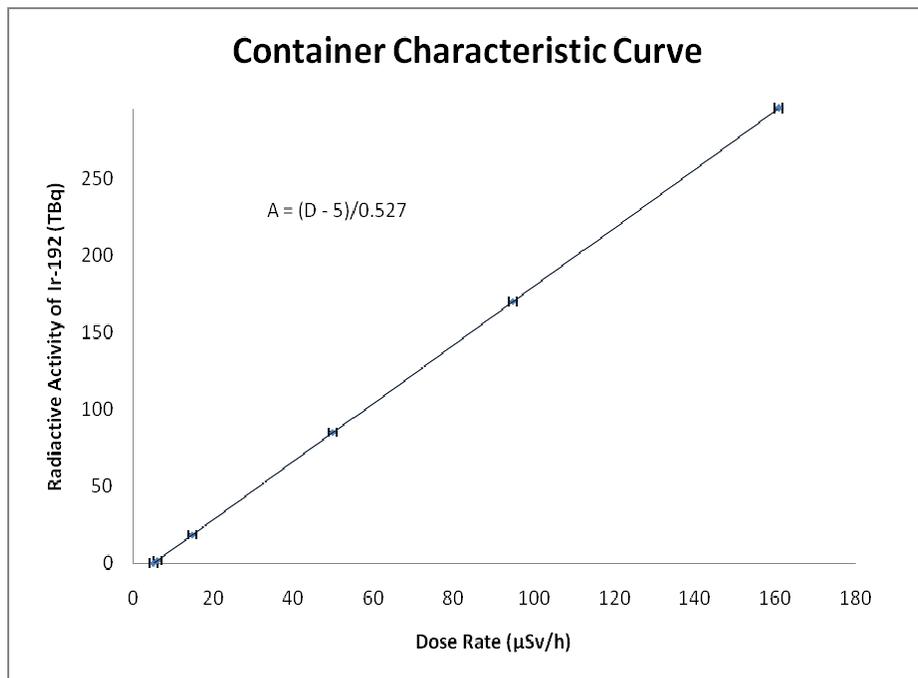
$$D_{total}^{\bullet} = 0.527 \cdot A + 5 \quad (6)$$

The aim of this work is to find a tool to estimate the radioactive activity of Ir-192 that contains a container from the measured dose rate at a certain point and outside the container. Therefore, we must invert equation (6) to obtain the radioactive activity of Ir-192 as a function of the total measured dose rate:

$$A = \frac{(D_{total} - 5)}{0.527} \tag{7}$$

This is the expression of the characteristic curve of the studied container that enables to theoretically estimate the activity inside the container by measuring the contact dose rate experimentally. The characteristic curve of the studied container is obtained according to figure 2.

Figure 2: Container characteristic curve



The points of this curve are representative and shows interval errors in both A and D_{total} .

The last point was taken in 296 TBq of radioactive activity and this is the maximum capacity that these containers can be transported.

3. Discussion

With the m value that was obtained, we were able to approximate the minimum detectable radioactive activity, considering the measurement error as mentioned in item “b” of table 1.

If we take 5 µSv/h as the dose rate that the shield gives, and that the measurement error is 1 µSv/h any dose above that value should be discriminated.

We can calculate the radioactive activity within our container for a dose rate of 1 µSv/h with the equation (7). The result is:

A	1.92 TBq
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From this result, we infer:

Each 1.92 TBq of Ir-192 to add inside the container corresponds to a dose rate of 1 μ Sv/h (in measuring instrument Geiger – Müller).

We know that the measurement error of $D_{\text{total}}^{\bullet}$ is $\pm 1 \mu\text{Sv/h}$ proceeds on the limit detection of Geiger-Müller detector as already mentioned. This dose rate value corresponds to an radioactive activity of 1.92 TBq.

Subsequently, we infer that the measurement error in the radioactive activity is $\pm 1.92 \text{ TBq}$.

As a result of this we can measure radioactive activities higher than 1.92 TBq of Ir-192 and 6 $\mu\text{Sv/h}$ is the minimum dose rate associated.

4. Conclusions

The regulatory control consists of the detection of Ir-192 sources within a container like the one that entered Argentina in March, 2007. The authors of the present work have confirmed that undeclared radioactive materials superior to 1.92 TBq of Ir-192 activity can be discriminated.

Our intention was to show that contact dose rate measured on the container in a given point could be used to quantify and control the radioactive activity declared using equation (7).

Especially in the case of imports or exports it would be useful to make an additional control when the radioactive material enters or leaves the country which allows to establish whether the radioactive activity declared by the sender is, with sufficient certainty, the one contained in the transport package. It would be extremely pretentious to establish a theoretical curve model for each and every model of container that may transport, besides, different radioisotopes. However, it would be adequate to count on the information since the construction of the different models in the case of transport of radioactive sources Category 1 and perhaps Category 2.

From the point of view of radiological safety it would be highly desirable to advance over the determination of such a model or even one of greater complexity to make quality controls and /or regulatory controls that allow to obtain a confirm of the radioactive activity .

There are different ways to make theoretical models and the one presented constituted the simplest approximation. However, it must be highlighted that it is much easier to do it empirically. Curve A versus D^{\bullet} can be constructed by taking experimental measurements in a certain point outside the package for different known values of radioactive activity of the radioisotope that is being studied and that cover the package spectrum of use. Mathematical models that take into account the design materials of those packages could also be used. This information could be part of the documentation concerning each package as the Certificate of Approval as type B (U) package.

The intention is to establish a curve for all the radioisotopes that the model of the package is able to transport and that the curve can be part of the certificate of approval of the package as such.

For this presented case, to count on this information would have been highly appropriate. To have made either quality and/or regulatory controls on time would have avoided the occurrence of the described event from the very beginning of the expedition.

Acknowledgements

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Individual Four Germanium Detectors Efficiencies for Uranium Lung and Lymph Nodes Burden Evaluation

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INDIVIDUAL FOUR GERMANIUM DETECTORS EFFICIENCIES FOR URANIUM LUNG AND LYMPH NODES BURDEN EVALUATION

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ABSTRACT

The nuclear regulatory authority possesses a system for actinide lung measurements. The lung counter is composed of a shielding chamber, four high purity germanium detector manufactured by Canberra called ACTII, and the corresponding electronic and software. Each detector has an active area of 3800 mm², a diameter of 70 mm and a thickness of 20 mm. They are placed inside a low background chamber built by the graded shielding technique, that consist (from exterior to interior) of 15 cm steel, 0.5 mm cadmium and 5mm lead. A Lawrence Livermore National Laboratory phantom was used to calibrate the system. Inactive organs and organs (lung and lymph nodes) loaded with a known amount of 235 uranium were used to quantify the response of the detectors. The spectra from each one could be analyzed and displayed separately or summed employing the GENIE-2000 software.

In a previous study by us the product of a simultaneous four-detector calibration was analyzed. In that opportunity, it was found the efficiencies and detection limits for different sets of active organs as function of the chest wall thickness. Within the analysis performed two aspects can be remarked. First, in practice, the calibration factors to be used are dependent on the real activity distributions with regarding that provided by the phantom. One of these discrepancies is the particular relationship between lung and lymph nodes activities present in each person that could be quite different from those of the phantom. Second, the detection limit of the system is high to be used for routinely monitoring. In this work, these features will be undertaken. Though, in order to get an improve estimation of the incorporated activity, it will be evaluated the possibility of using the individual calibration and measurements of each detector in a series of equations to determine the lymph nodes-lung activities relation presented in a real situation. A comparison of the efficiencies obtained for each detector and an analysis to find an optimal combination of them for activity detection will be performed. To begin analysing the second subject, the feasibility of using summing techniques to reduce the detection limit will be also evaluated. To perform this task, spectrums from voluntaries persons will be employed.

KEYWORDS: *lung, lymph nodes, calibration, uranium, detection limit, efficiency.*

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Individual four germanium detectors efficiencies for uranium lung and lymph nodes burden evaluation

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INTRODUCTION

It is known that a good assessment of the retained activity will depend on how similar the calibration conditions were to those that appeared in the real circumstances. Among the elements with a significant influence on the assessment are the differences in the anatomical structure and activity distributions with regarding to that provided by the phantom. In a previous study of the Nuclear Regulatory Authority (ARN) lung counter system, the product of a simultaneous four-detector calibration was analyzed⁽¹⁾. In that opportunity, it was found the efficiencies and detection limits for different sets of active organs as function of the chest wall thickness. Within the analysis performed one aspect can be remarked, in practice, the calibration factors to be used are dependent on the real activity distributions relationship between lungs and lymph nodes activities present in each person that could be quite different from those of the phantom. Though, in order to get an improve estimation of the incorporated activity, it was evaluated the possibility of using the individual calibration and measurements of each detector in a series of equations to determine the lymph nodes-lung activities relation presented in a real situation. A comparison of the efficiencies obtained for each detector and an analysis to find an optimal combination of them for activity detection was performed.

(ARN) lung counter system: This is composed of four high purity germanium detector with an active area of 3800 mm², a diameter of 70 mm and a thickness of 20 mm. They are placed inside a low background chamber built by the graded shielding technique, that consist (from exterior to interior) of 15 cm steel, 0,5 mm cadmium and 5mm lead. A Lawrence Livermore National Laboratory phantom was used to calibrate the system. Inactive organs and organs (lung and lymph nodes) loaded with a known amount of ²³⁵U were used to quantify the response of the detectors.

RESULTS

The results presents in this work are limited to the case of 185,72 KeV of ²³⁵U and an effective chest thickness of 2,958 cm.

Efficiency: The detectors positions over the chest phantom are shown in Fig. 1. In Fig. 2 and 3 it is presented the individual organs (lung or lymph nodes) efficiency factor obtained for each detector. With them a total efficiency given by the Eq.(a) can be obtained.

$$Eff_{total} = \frac{Eff_{lungs} + x Eff_{nodes}}{1+x} \quad \text{with } x = \text{Nodes Activity/Lung Activity} \quad (a)$$

This formulae evidences the necessity of knowledge the particular relationship between lymph nodes and lungs activities. The real x factor value is usually impossible to determine. A way to refine this parameter, if the time of intake is known, could be the employment of values given by the metabolic standard models. In other case, a technique to approximate x could get through the individual calibration of each detector for the different sets mentioned above. This information and the measured areas of the interest photopeak given by each detector will generate the following series of equations:

$$\frac{M_i}{Eff_{nodes_i}} = \frac{Eff_{lungs_i}}{Eff_{nodes_i}} A_{lungs_i} + A_{nodes_i} \quad 1 \leq i \leq 4 \quad (b)$$

where M_i is the photopeak area and A are the nodes and lung activities given by the detector i .

With the constrain that the lung and lymph nodes activities determined by each detector must be similar, equation (b) will correspond to a straight line whose parameters (and in consequence x) might be obtained through least mean square techniques.

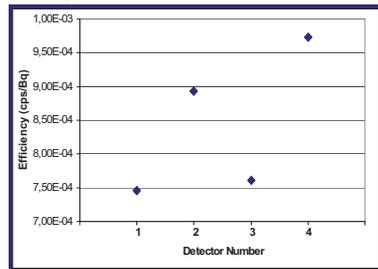


Figure 2. Detector efficiencies for lymph nodes.

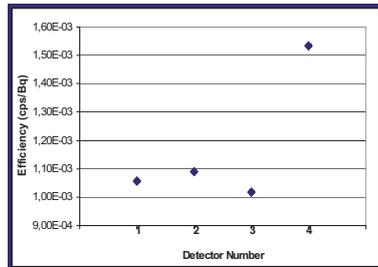


Figure 3. Detector efficiencies for lungs

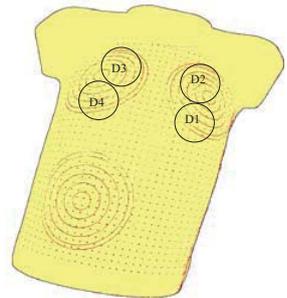


Figure 1. Detector positions.

Lymph nodes and lungs activities estimations: According to the activities reported in various publications the x factor could be ranging from 0.05 to 36.14⁽²⁾. This implies that the activity of the lymph nodes could vary from 5 % to approximately 3600% respecting to the lung activity. Taking into account this values, spectrums corresponding to different activity percentages were simulated and, using the individual detector efficiencies and Eq. (b), the individual lungs and lymph nodes activities were tried to reproduce for diverse sets of detectors. Table I shows the differences obtained among the lung and lymph nodes activities and those estimated through Eq. (b).

Table I								
Percent differences between lungs activity and activity estimated via the detector spectrums								
Percentage of lymph nodes activity as function of lungs activity	Numbers of the detectors in use							
	1, 2, 3 and 4	1 and 2	3 and 4	2 and 4	1, 2 and 3	1, 2 and 4	1, 3 and 4	2, 3 and 4
5	-3	-9	-15	<-1	<-1	<-1	-11	-2
25	-6	-33	-36	5	-23	3	-23	-1
50	-4	-21	-25	2	-14	1	-17	-2
75	-4	-16	-22	1	-11	1	-15	-2
95	-5	-25	-29	3	-17	2	-19	-1
105	-5	-26	-30	3	-18	2	-20	-1
200	-4	-11	-17	0	-7	0	-12	-2
500	45	-21	259	2	-61	1	225	39
Percent differences between lymph nodes activity and activity estimated via the detector spectrums								
Percentage of lymph nodes activity as function of lungs activity	Numbers of the detectors in use							
	1, 2, 3 and 4	1 and 2	3 and 4	2 and 4	1, 2 and 3	1, 2 and 4	1, 3 and 4	2, 3 and 4
5	96	209	466	-6	155	-8	328	76
25	44	74	92	30	66	30	69	41
50	-26	-9	6	-35	-14	-35	-8	-28
75	-49	-36	-22	-57	-40	-57	-34	-51
95	7	31	48	-5	24	-5	30	4
105	6	29	46	-5	23	-6	28	4
200	-78	-70	-58	-94	-74	-84	-66	-79
500	-107	-9	-419	-35	24	-35	-361	-102

Note: A negative difference indicates that the activity is underestimated

In table I it was not presented the detectors combinations (1 and 3), (1 and 4) and (2 and 3); for the first group, the differences were superior to 100% in most situations and for the other two the discrepancies were superior to 50% nearly all for the lymph nodes and in a few cases superior to 100%. For lung the differences remain between 15 and 88%.

From the table, it can be seen that when the detectors 2 and 4 were used simultaneously the differences in the estimation of lungs activities are inferior to 5% and in most cases the deviations in lymph nodes activities are inferior to 50%.

CONCLUSIONS

In all situations the estimation of lungs activities had the minor error compared with that obtained for lymph nodes. They were in a good number of the conditions studies (different detectors combinations or lymph nodes-lungs activities relationships) lower than 30%, decreasing to 5% if the detectors 2 and 4 are simultaneously in use. For lungs the disagreements are predominantly between 6 to 50%. Considering that all the detectors grouping must conduct to the same value of activities estimations, it could be selected several of the most favorable combinations and try to find with them the x factor; but to establish a generalization for the system this study is going to be extended to new activities distribution and CWT.

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Argentina, Regional Training Center on Radiation Protection for Latin America

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Argentina, Regional Training Center on Radiation Protection for Latin America

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Abstract

Argentina has an extensive background in education and training on Radiation Protection. Since the beginning of the nuclear activity in the country, prominence was given to the aspects related to radiation protection and training of the personnel involved in the use of ionizing radiation. These educative activities have been delivered for more than 50 years, having accumulated an important experience in the field.

The Nuclear Regulatory Authority has the statutory obligation to address, among other matters, the control of the aspects of nuclear safety and radiation protection on the whole country, to protect the people of the harmful effects of ionizing radiation resulting from the nuclear activities. This includes the responsibility to develop and enforce the regulations, standards and other requirements, particularly, establishing the requests and promoting activities regarding education and training on radiation protection.

Argentina, currently through the Nuclear Regulatory Authority, has performed postgraduate courses on radiation protection and nuclear safety at interregional and regional level for 28 years without interruption. This important experience has been valued and exploited to form a Regional Center on Education and Training for Latin America and the Caribbean, sponsored by the International Atomic Energy Agency.

The Regional Center that in fact has been running in Argentina, trained 404 foreign participants and 327 local participants since 1980, totalizing 731 graduates from our annual post graduate courses. Our commitment is that all the effort made in education and training on radiation protection and nuclear safety contributes to a better use of the benefits of nuclear development.

Since 2001 the International Atomic Energy Agency raised the need to develop plans and establish agreements to ensure a long-term sustainability of the education and training programs, allowing a better use of the resources in this area. In order to achieve this goal, Regional Centers for Education and Training were established in different geographic areas and languages.

According to that strategy, Argentina, through the Nuclear Regulatory Authority, will continue working in a systematic way to disseminate the latest knowledge and experience on radiation protection, in Spanish language, to the whole region of Latin America and the Caribbean.

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INTRODUCTION

1.1. History

Nuclear activity in Argentina began in the year 1950. The leader institution in this area was the National Atomic Energy Commission (CNEA in Spanish) created through the Decree 10.936/50 dated on May, 31st 1950.

The Act 22.498/56 established in 1956 the legal structure of the CNEA and defined in the Article 2 “the responsibility of the new Institution in the control of scientific and industrial applications of the transmutations and nuclear reactions for reasons of public convenience or to prevent damages that they could cause”.

To comply with the Article 2 mentioned above, the Board of Directors of CNEA approved in 1958 the “Regulations for the Radioisotopes and Ionizing Radiations Use”. It came into force the 24th of January, 1958 by Government Decree 842/58.

In Article 19 the Decree mentioned that every person interested in the use of radioisotopes shall:

Have the knowledge and experience, acquired in a national or foreign specialized centre, fully documented, to use the specific radioisotopes, along with the corresponding on the job training for at least one year, or:

1. Have completed, a national or foreign specialized theoretical and practical course about the use of radioisotopes, with the corresponding examinations, enough to qualify the participant to handle the specific radioisotopes. That course must include concepts related to radiations physics, radioactivity, radiochemistry, measuring instruments, radiology health physics and radiation protection. Its duration must be at least 50 hours-class including theoretical and practical activities.
2. Have performed professional practice for the use of the radioisotopes that the participants wish to apply, in an authorized centre, according to the rules that for each case the CNEA will deliver.

It means that five months earlier than the *United Nations Scientific Committee on the Effects of Atomic Radiation* (UNSCEAR) approval of its first report (June 13th 1958), almost a year before the first publication of the *International Commission on Radiological Protection* (ICRP) (dated by the end of 1958) and when the International Atomic Energy Agency (IAEA) was starting to function, Argentina already had regulation, approved by the corresponding authority, that included specifically the training needs in radiation protection.

However, an important issue of radiation protection regulation was still absent by that time. The functions assigned by law to the CNEA excluded the control of x rays generating equipment, because they were under the surveillance of the National Ministry of Health. This situation made some CNEA professionals to promote a specific law for the use and control of those radiation sources, assigning the responsibility to the corresponding National and Provincial authorities.

As a result of this initiative was promoted the Act N° 17.557/67, known as the “X Rays Law”, in 1967. This law established the complete regulatory scheme for ionizing radiation applications in Argentina, which is still in force today. It happened at a very early stage, 10 years before the Publication 26 of the ICRP.

During the 90`s another important fact contributed to strengthen the regulatory framework for the control of ionizing radiation use in the country:

In 1994, Presidential Decree 1540/94, split from CNEA the area of radiation and nuclear safety control, constituting a new independent organization initially named “*Ente Nacional Regulator Nuclear*”.

Then in 1997, was approved the Act N° 24.804/97 known as the “National Nuclear Activity Law”, with its corresponding explanatory Decree N° 1.390/98. After that the control organization changed definitely its name to Nuclear Regulatory Authority (ARN in Spanish).

The ARN might be considered a new institution, but it has personnel and organizational resources with almost 50 years of experience as the CNEA regulatory branch. It means that it can be considered close to Phase III (maximum development level) according to IAEA standards.

1.2. Legal Organization

In the Argentine Republic, the ARN has the mission of, inter alia, protect the people against the negative effects of the ionization radiations coming from nuclear activities (Act N° 24.809/97). The Ministry of Health and Environment (MSA in Spanish) is in charge of people’s protection through the control of the x-rays equipments. (Act N° 17.557/67)

Both Institutions work complementarily to regulate the whole utilization of ionization radiations in the national and federal level.

In the area of x rays regulation, the legal instruments (act and decrees) determine the training needs for the education and training in radiation protection. These courses are delivered by personnel of the MSA and the Provincial Health authorities

The regulation for the training in radiation protection of the licensees, under the ARN area, is more complex. It depends of the risk level of each installation or equipment, as it is defined in the Basic Safety Standard (AR 10.1.1) approved by the ARN.

This scheme classifies the installations in three types or classes that in order of decreasing risk are:

Class I Installation: Installation or practice that requires a licensing process of more than one step.

Class II Installation: Installation or practice that only needs an operation license.

Class III Installation: Installation or practice that only needs to be registered.

2. EDUCATION AND TRAINING IN RADIATION PROTECTION

2.1. National Program

In Argentina, there is in fact a national program for education and training in radiation protection and safety. It is organized through a vast set of courses and degrees that bring the corresponding theoretical knowledge, and institutions that complete the on the job - training needed. All this educational system is led by the ARN and the MSA.

Both institutions, as the authorities of regulation and control, manage independently the specific subjects of their interest, like education and training of human resources. Nevertheless, for some topics there are coordination and working mechanisms between the two authorities.

The MSA establishes the knowledge requirements for people that would apply for the x ray equipments use. The ARN rules the theoretical and training requirements for licenses in the area of ionization radiation from nuclear activities.

There are specific laws, decrees and regulations that specify the minimal requirements in theoretical knowledge and on the job training that must be reached for a safe use of ionization radiation.

The following information shows the education and training required by the ARN to all the personnel that work in licensed positions in Class I, II and III installations. (Regulation AR 10.1.1)

There are three basic levels of training required by the ARN:

- **Basic knowledge:** the academic grade studies required as a minimum for a regulated position.
- **Specific knowledge:** the specialized courses or degree education and training required for the application to a particular practice or job position.
- **On the Job Training:** the practical work during a precise time under the supervision of an expert.

2.2. Education and Training Activities

The ARN and its related institutions organize more than thirty training events annually for around 600 participants, being by far the main training providers at the national level. As the main regulatory authority in the country, ARN has regulatory control over two NPPs in operation, one under construction, three critical assemblies, three research and isotope production reactors, 25 major radioactive facilities and more than 1600 facilities using ionizing radiation for medical, industrial, research or training purposes.

The ARN delivers its own courses, but besides that, encourages and supports other professional organizations and scientific institutions to dictate different degrees or courses, necessary to fulfill the specific educational level that is required to apply for permissions or licenses in the nuclear area.

The ARN also has as a permanent policy to organize, deliver and assess Train The Trainer courses (TTT), as a way to increase the teaching abilities of the professionals in charge of the lectures, and allow new teachers to multiply and spread the educational effort.

The next Tables show the courses that the ARN and the MSA deliver and those that have been recognized by the ARN as part of the human resources training program for different practices with ionizing radiation.

TABLE I COURSES DELIVERED BY THE NUCLEAR REGULATORY AUTHORITY AND THE FEDERAL MINISTRY OF HEALTH			
COURSE	TRAINEES	AVERAGE PARTICIPANTS PER YEAR	LEVEL
Postgraduate Educational Course on Radiation Protection and Safety of Radiation Sources.	RPO in nuclear and radioactive installations candidates; Regulatory authorities new personnel.	23	G
Postgraduate Educational Course on Nuclear Safety.	Nuclear facilities personnel; regulatory authorities personnel.	15	G
Radiation Protection for Technicians	Personnel in nuclear and radioactive installations and RPO candidates.	23	UG and G
Regional Educational Course on Physical Protection of Installations and Nuclear Materials.	Nuclear installations operators	24	G and UG
Regional Educational Course on Safe Transport of Radioactive Material.	Federal regulatory and control branches and Transport operators	15	G and UG
Regional Educational Course on Radiation Emergency Medical Preparedness	Medical practitioners and RPO.	25	G and UG
Biological Effects of Ionizing Radiation and Emergency Medical Preparedness.	NPP personnel	30	G and UG
Regional Educational Course on Prevention of Illicit Trafficking of Nuclear and Radioactive Materials.	Security Forces; Intelligence officials; Customs personnel; Internal Revenue personnel; Foreign Trade university students	1700 (2001 to 2008)	G and UG
Health Physics (Radiation protection in practices with x rays)	Medical practitioners (radiologists, surgeons; hematologists) Dentists; Engineers; Technicians; Customs and Security personnel	1500	G and UG
Health Physics (Radiation protection in practices with x rays)	Medical practitioners (radiologists) Dentists; Veterinaries; Engineers; Physicists; Operation and maintenance technicians; Security Forces	1525	G and UG
Health Physics (Radiation protection in practices with x rays)	Customs and Security Forces personnel	380	G and UG

TABLE II COURSES RECOGNIZED BY ARN OR FEDERAL MINISTRY OF HEALTH				
PURPOSE	COURSE	TEACHING ORGANIZATION	AVERAGE PARTICIPANTS PER YEAR	LEVEL
Theoretical knowledge and clinical practice in Nuclear Medicine	Postgraduate Course for Nuclear Medicine Specialists	University of Buenos Aires, School of Medicine and Atomic Energy Commission	7	G
Medical use of Radioisotopes Training	Radionuclides Applications Methodology	Nuclear Studies Institute. Atomic Energy Commission	8	G
Medical use of radioisotopes technicians training	Nuclear Medicine Technicians	Nuclear Studies Institute. Atomic Energy Commission	7	UG
Medical Use of	Nuclear Medicine Technicians	Hospital Juan A. Fernandez – Nuclear Medicine Unit	10	UG

TABLE II COURSES RECOGNIZED BY ARN OR FEDERAL MINISTRY OF HEALTH				
Radioisotopes Technicians Training				
Undergraduate Nuclear Medicine Technologist Education	Imaging Diagnostic Systems Technologists	San Martin University.	13	UG
Postgraduate Training in Medical Applications (Radiotherapy)	Radiotherapy Dosimetry	Nuclear Studies Institute. Atomic Energy Commission	12	G and UG
Postgraduate Training in Radiotherapy Physics	Radiotherapy Physics Specialist	Nuclear Studies Institute. Atomic Energy Commission	12	G
Medical Physics Education	Medical Physics	San Martín University.	3	UG
Medical Physics Education	Medical Physics Engineering	Favaloro University.	12	UG
Postgraduate Medical Physics Education	Medical Physics	University of Buenos Aires School of Sciences	-	G
Postgraduate Medical Physics Education for Qualified Experts in Radiotherapy Physics	Master Science in Medical Physics	Balseiro Institute Nuclear Medicine School (Mendoza)	8	G
Postgraduate training in research and teaching	Radiochemistry and Nuclear Chemistry	University of Buenos Aires, School of Sciences (INGEIS) -	-	G
Education and Training in Nucleonic Gauges	Radiation Safety in industrial use of sealed sources	South University – Radioisotopes Laboratory	10	UG
Education and Training in Nucleonic Gauges	Nucleonic gauges use	Radiation Protection Society	78	UG
Education and Training for Gamma Radiography operators	Gamma Radiography permits	Non Destructive Testing Centre	52	UG
Education and Training for Gamma Radiography Operators Permit Renewal	Gamma Radiography Refresher course	Non Destructive Testing Centre	12	UG
Education and	Gamma Radiography	Technological University (Mendoza)	-	UG

TABLE II COURSES RECOGNIZED BY ARN OR FEDERAL MINISTRY OF HEALTH				
Training for Gamma Radiography Operators Permit Renewal	Refresher course	Campus) – Regional Technological Testing Institute		
Education and training for Gamma Radiography Operators Permit Renewal	Gamma Radiography Refresher course	Radiation Protection Society	8	UG
Education and Training in RIA.	Radioisotopes Applications Methodology	University of Buenos Aires – Biochemistry and Pharmacy school	14	G
Education and Training in RIA.	RIA	Biology and Experimental Medicine Institute	-	G
Education and Training on Radioactive Tracers in Oil Industry and Well Logging.	Radiological Safety in Radioisotopes Applications in the Oil Industry	NOLDOR S.R.L.	-	G and UG
Nucleonic Gauge Permits.	Radiological Safety applied to Nuclear Instruments of Measurement and Control for Industrial Use	NOLDOR S.R.L	17	UG
License Application for Mobile Irradiators	Training of personnel in mobile irradiators	Nuclear Medicine school (Mendoza)	12	UG
License Application for Cyclotron and PET	Training of personnel in Cyclotron and PET	Nuclear Medicine school (Mendoza)	8	G and UG
License Renewal for Cyclotron and PET	Refreshing course on radiation safety in cyclotron and PET	Nuclear Medicine school (Mendoza)	7	G and UG
Education and Training on Radioactive Tracers in Oil Industry and Well Logging	Radioactive Tracers in Oil Industry and Well Logging	Argentine Radiation Protection Society	8	UG
Education and Training on the Use of Low Activities Radiation Sources	Import. Export and Sale of radioactive material. Use of low activity radiation sources	Argentine Radiation Protection Society	60	UG
Health Physics (Radiation protection in practice with x ray)	Medical practitioners (radiologists) Dentists; Veterinaries; Radiology technicians.	Nuclear Medicine School (Mendoza)	240	G and UG

TABLE II COURSES RECOGNIZED BY ARN OR FEDERAL MINISTRY OF HEALTH				
Education and Training for License Application (basics)	Basic course on Radiation Protection (recognition in process)	Atomic Energy Commission Bariloche Atomic Center	15	G and UG
Postgraduate Medical Physics Education for Qualified Experts in Nuclear Medicine Physics	Master Science in Medical Physics (recognition in process)	Balseiro Institute Nuclear Medicine School (Mendoza)	7	G

The national program for education and training in radiation protection and safety, mentioned above, determines minimum requirements for each practice with ionizing radiation and also the mechanisms to verify its correct development.

For the x rays area, under the regulation of the MSA, the fulfillment of the education and training standards is direct because the Ministry is the only institution that delivers the courses and the corresponding evaluations, following its own quality assurance program.

The ARN has recognizing mechanisms to accept and audit the different courses, grade careers, and training centers to certify that the courses comply with the requirements of the regulatory framework.

The next Table completes the general view of the national program for education and training in radiation protection and safety with some courses and degrees in which radiation protection and safety are included in their syllabus. Note that these examples do not follow a regulatory requirement.

TABLE III: DEGREES AND COURSES THAT CONTAIN RADIATION PROTECTION IN THEIR CURRICULA	
DEGREES/ COURSES	TEACHING ORGANIZATION
Nuclear Engineering	Balseiro Institute Cuyo University
Nuclear Energy Technological Applications	Atomic Energy Commission (CNEA) Cuyo University (Balseiro Institute) University of Buenos Aires (School of Engineering)
Master Degree in Nuclear Reactors	Atomic Energy Commission (CNEA) National Technological University (UTN)
Master Degree in Radiochemistry	University of Buenos Aires (School of Sciences)

TABLE III: DEGREES AND COURSES THAT CONTAIN RADIATION PROTECTION IN THEIR CURRICULA	
Medicine	University of Mendoza (School of Medicine)
Bio image Production	University of Civil Marine (School of Engineering)
Medical Engineering Specialist	Favaloro University National Technological University - San Nicolas
Nuclear Techniques Specialization Course	Atomic Energy Commission (CNEA) Bariloche Atomic Centre

3. PGEC AND THE REGIONAL TRAINING CENTRE

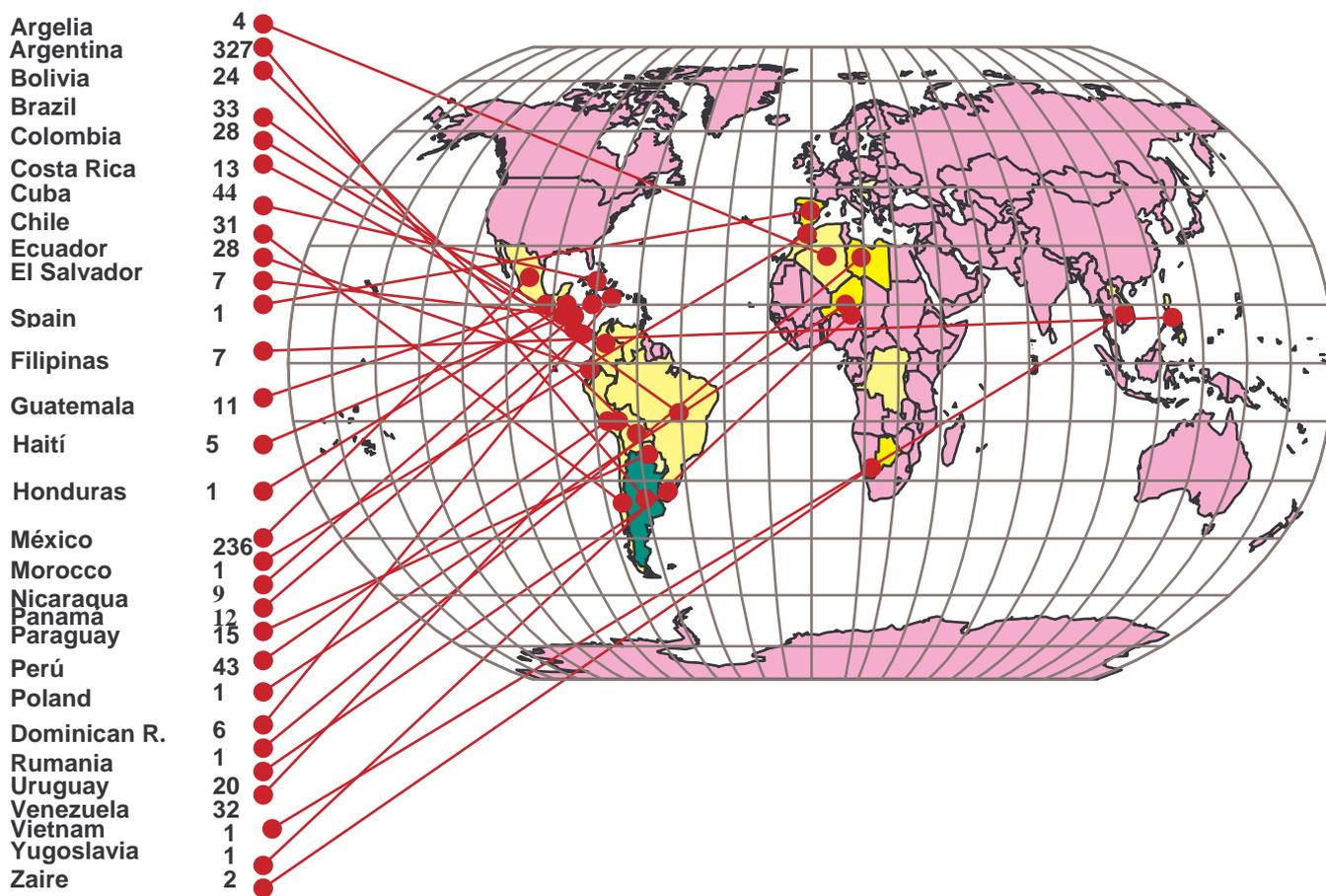
3.1. Antecedents of the Post Graduate Educational Course (PGEC) in Argentina

Argentina, currently through the Nuclear Regulatory Authority, has performed postgraduate courses on radiation protection and nuclear safety at interregional and regional level for 28 years without interruption.

The Postgraduate Educational Course on Radiation Protection and Safety of Radiation Sources (PGECRPSRS) and the Postgraduate Educational Course on Nuclear Safety (PGECNS) are delivered every year. The first with approximately 125 days duration from March to September, and the second with approximately 60 days duration from September to December. Both courses have a significant participation of representatives from the Latin-America region, with around 15 and 8 foreign participants each year respectively. The following graphic shows the origin of participants along years.

This important experience has been valued and exploited to establish a Regional Centre on Education and Training for Latin America and the Caribbean, sponsored by the International Atomic Energy Agency.

Map 1: ASSISTANTS TO THE POST GRADUATE COURSES ON RADIATION PROTECTION AND NUCLEAR SAFETY



PERIOD FROM 1980 TO 2008: TOTAL 731

3.2. Education and Training Appraisal (EduTA)

Argentina has carried out a self appraisal process regarding its infrastructure in the field of National Education and training, following the guidelines stated by the International Atomic Energy Agency in a document that is nowadays an advanced draft named “*Education and Training Appraisal in Radiation Protection and the Safety of Radiation Sources*” (EduTa). The developed process included in 2006 the invitation to an international mission for visiting the country in order to observe the organization and training capabilities related to radiation protection and radiation sources safety.

EduTA Document provides the appraisal guidelines. It covers every institution, organization, facilities, equipment, personnel and documentation related total or partially with education and training in radiation protection and radiation sources safety.

The appraisal pointed out in the document is applicable, among others, to the Member States with capacity to provide regional training courses or post-graduated courses promoted by the IAEA.

For those countries that provide courses, this document offers a methodology for evaluating Post-graduate courses on radiation protection and radiation sources safety (PGEC) and the course for radio protection officers (RPO).

EduTA is thought to be applied in two ways:

- Appraisal by an expert group of the IAEA, or
- Self- appraisal by the Member State.

Argentina decided to carry out a self-appraisal process, which results were subsequently put to a mission of international experts coordinated by the IAEA to judge.

3.3. Argentine’s Self-Appraisal Process Output

The first main stage, named *pre-appraisal*, took place during the last quarter of the year 2005. It was carried out by a team of professionals of the Nuclear Regulatory Authority and the Ministry of Health and Environment.

The work demanded an assessment of the activities directly or indirectly related to training in radiation protection over the entire country.

The final results of the pre-appraisal are found in the document edited by the ARN “*Education and Training in Radiation Protection and the Safety of Radiation Sources. Pre-Appraisal Information*” Nuclear Regulatory Authority, March 2006.

The second stage, named *appraisal*, was carried out since the end of the previous one to the mid of 2006. The same team of professionals that had worked previously developed it.

The main topics raised were:

- Assessment of the regulatory infrastructure for education and training
- Assessment of the national strategy to establish education and training competences
- Assessment of the national needs in education and training
- Assessment of the infrastructure for education and training
- Assessment of the Post-Graduated courses promoted by the IAEA
- Assessment of the training courses useful for radio protection officers

In May 2006 the *appraisal* was finished. As a result of the mentioned evaluation process it was observed that Argentina satisfies almost all the requirements stated by the IAEA in its document EduTA.

The arrived results and conclusions, that also included the points of view of the work team, regarding the document followed to carry out the evaluation, were exposed to an expert's mission of different countries coordinated by the IAEA that visited Argentine.

3.4. Regional Training Center for Latin America and the Caribbean.

Once approved and published the results of the appraisal process the country was officially recognized by the IAEA as Regional Training Center on Radiation Protection for Latin America and the Caribbean. Thus, Argentina is responsible of providing technical courses, degrees and Post-graduated courses, promoted by the IAEA in Spanish language.

A long term compromise between Argentina and the IAEA is predicted during the year 2008. This is part of the Strategic Plan stated by the Agency for the period 2001-2010 according to the resolutions of the General Conference GC(44)/RES/13 of 2000, GC(45)/RES/10C of 2001 and GC(49)/RES/43, that encourage the establishment of this kind of agreement between the IAEA and the Regional Training Centers.

4. REGIONAL TRAINING CENTERS

4.1. The IAEA Strategic Plan for Education and Training

Since 2001 the International Atomic Energy Agency raised the need to develop plans and establish agreements to ensure a long-term sustainability of the education and training programs, allowing a better use of the resources in this area.

In order to achieve this goal, Regional Centers for Education and Training are being established in different geographic areas and languages.

COUNTRY	REGION	YEARS DELIVERED	LANGUAGE
ARGENTINA	LATIN AMERICA	28	SPANISH
BELARUS	EUROPE	5	RUSSIAN
GREECE	EUROPE	3	ENGLISH
MALAYSIA	ASIA	5	ENGLISH
MOROCCO	AFRICA	5	FRENCH
SOUTH AFRICA	AFRICA	1	ENGLISH
SYRIA	ASIA	7	ARABIC

The following map shows the worldwide distribution of the Regional Training Centers.

Map 2: IAEA REGIONAL TRAINING CENTERS



Source: IAEA Web - ICN

4.2. Long Term Agreements

The long term agreements will remain for a negotiable period between the IAEA and the countries which act as headquarters of the Regional Training Centers. The objective is to give continuity to the policies developed in education and training in the different regions of the world.

The agreements entail a commitment from the two sides with the development of the training activities in radiation protection and must specify:

- Type and frequency of the training events to develop in the host country (education courses for post graduated, pedagogic courses for trainees, courses for radiation protection officers, etc)
- Relevant subjects according to the region
- Necessary budget for training activities and participants support
- Monitoring of the participants performance during the courses in order to assure the training effectiveness

5. CONCLUSIONS

As part of the education and training in radiation protection strategy, Argentina, through the Nuclear Regulatory Authority, will continue working in a systematic way to disseminate the latest knowledge and experience on radiation protection, in Spanish, to the whole region of Latin America and the Caribbean.

Argentina, that has a long experience in this field, considers that the safe use of the benefits that atomic energy offers in its diverse applications entails the devotion of important resources, experience and dedication for the education and training in radiation protection and personnel safety.

In relation with this conviction is that the country imposed itself to plan and carry out a suitable diffusion of knowledge and experience in Spanish language over the entire region, taking better advantage of the available resources.

Radiation Safety and Culture of Prevention in the Use of Radioactive Materials in Industry. Criteria and Trends

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Radiation safety and culture of prevention in the use of radioactive materials in industry. Criteria and trends.

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Abstract. As time goes by and experience is gained, modernization and technological development show the need to implement more complex programs and procedures to ensure a high level of compliance with radiation safety, particularly in those activities in which radioactive material is used in industry. A relevant aspect of present technology is the concern to introduce mechanisms to prevent radiological accidents or incidents, to ensure early detection of failures. This includes systems that either individually or as a whole, increase the level of responsibility of the different disciplines involved, so as to avoid a situation that could lead to loss of control of the facility or part of it. The prevention of an abnormal situation, overexposure of workers or unwanted risks, should be considered in the level of vulnerability of the facility, a concept drawn from international protection systems and which is applied directly in radiation safety. Preventive management, risk communication and proposals for change or improvement along with the detection of risks and training, constitute all the factors contained within prevention policies. Dose limitation, optimization and justification, old tools used for decades, could not be replaced by other modern concepts and criteria. ALARA culture (including performance indicators) should be considered. The atmosphere at work, working under pressure as well as other factors such as quality issues, ethics of prevention, etc. align with this idea of prevention and safety, besides changes in attitude, towards risk prevention (methods, reports, intervention guides, working instructions, and any other helpful tool), are followed by preventive, as well as predictive and corrective maintenance, applied to minimize the dose absorbed by workers. A clear policy of prevention is needed as well as an appropriate level of radiation safety which should be taken into account since the very beginning of the development of a given practice. All these concepts are being described in this paper.

KEYWORDS: *Radiation, safety, prevention, industry.*

Introduction.

In recent years, incidents or accidents have been reported to international information systems for dissemination and knowledge (NEWS -Nuclear Events Web Based System). From the information received is detected in general, situations arising from ignorance or misinformation, lack of "perception of radiological risk" absence of procedures, etc., by the user of radioactive material. This aspect, it is not minor at the time of making the right decision, once detected the incident or radiological accident, exposes the need to develop a plan contingencies to remedy quickly and efficiently the consequences of the act.

This means that a poor culture as it relates to the perception of risk will result in a complication even more time trying to resolve the emergency preparedness and necessary care. [1][2]

The risks of radiological accidents, which are associated with the tasks using radioactive materials, in whatever form, although its rate of occurrence is low, are mistakes when designing the installation, during the same operation, or some practice process defined or poorly studied, and so on. whose characteristics depend on the analysis and strategy developed when deciding its implementation. In them, abuse, maintaining performed by staff without the knowledge, failures or oversights of human kind and incidents identified the magnitude of the event. All these "factors" determine the need to implement aspects of study and analysis, as well as early detection of failures and the adequacy of implementation procedures.

Description.

The handling of radioactive material leads to the operator exposition but the received doses in normal operation are below the annual limits set in the legislation in force both in the National as well as the International level. These values are optimized to guarantee even a lower exposition, constituting in this way a preventive politics. [3][4]

Radiological protection applied to exposed personnel must introduce improvements and avoid unjustified or non desired doses. It also must guarantee the physical protection of radioactive material,

taking the corresponding control measures and trying to modify the habits and behavior of the personnel occupationally exposed towards safety culture and prevention. This is the reason why it is necessary to set the objectives, to select adequate protection systems according to the kind of practice so as to be able to act immediately and in a safe way both in a normal or an abnormal situation.

Due to the natural evolution of the basic principles of protection, the optimization and the control mechanisms used, we can establish adequate level of restrictions with the consequent benefit for the worker.

All these concepts used must be introduced in every application of the radioactive material in industry. In this regard Nuclear Regulatory Authority of República Argentina (NRA) has required improvements in industrial facilities through the addition of tools that allow ensure a higher commitment with radiological safety and prevention. The inclusion of certain issues related to, for instance, preventive management, risk communication and other concepts is relevant to achieve the evolution of the regulatory system, according to the kind of risk involved in the use of radioactive material.

In this way the NRA has implemented the analyses of deeper regulatory standards as well as the inclusion of safety culture aspects by introducing quality assurance in an attempt to guarantee prevention culture concepts.

Concepts about improvements and methodologies to be applied in order to get an adequate level of safety during the development of a certain practice with radioactive material are described. [5]

1) Vulnerability.

The methodology used to determine the vulnerability of a given system is based on the determination of the negative consequences for the process and the structures that are exposed to a risk factor. Several methods exist at present and by means of the application of these models an equation that involves certain risk and the probability of its occurrence in time is determined through studies or tree failure.

In this way it is possible to predict what will happen in case of an accident or incident. It will be possible to evaluate, in a theoretical way, vulnerability models that allow for example to determine in what way a given material will disperse into the atmosphere in case of an uncontrolled escape or in case of an explosion.

2) Preventive management.

Radiological safety principles establish the necessity of taking into account safety among the workers, members of the public and the environment when operating. This is the reason why the use of procedures, guides and regulation that lead to accomplish a high level of protection is very important. Even when the principles of use of a given practice are determined beforehand is eventually the operator or user who can notice a defect or mistake in the first place when a certain procedure is being used. This will allow the adoption of practical solutions and the implementation of an adequate level of safety for it will be the operator who will be in charge of suggesting the necessary changes in order to guarantee the correct use of methods or systems that use radioactive material demonstrating a high level of compromise with safety and generating a positive attitude towards improvement and prevention what turns him in an independent examiner. [6]

3) Risk communication.

Regarding risks communication, the lack of this basic principle on the part of the facility may result in complicating a task that was easily controllable at the beginning. In many cases the material involved may present anomalies, alterations in the way of performing the tasks, modifications in technologies etc. which can be detected and corrected if the risks were clearly defined at the moment of determining the process or technique to use.

Applying radiological protection principles is a basic condition for prevention and it must be considered a severe fault not to comply with all procedures and set safety guides, performing the necessary safety control audits to verify they truly are complied with. [7][5]

Uncontrolled or accidental situations are failures related to management, consequently they are avoidable and emphasis must be made in complying with the level of safety that the practice demands, independently of the degree of complexity and duration. For that purpose basic principles of

radiological safety together with operative requirements that imply a high level of compromise on the part of the operator and the installation and safety guides must definitely be established.

4) Improvement and change proposals.

It is relevant to analyze development, modernization and the application of conducts oriented to detect risks presented by process of changes due to the use of modern technologies, the updating of procedures, quality improvements or adaptation to new regulations and the impact on operators and the facilities. This process has negative and positive aspects. (Table 1)

Training in attitude cannot be reached overnight. The personnel responsible must establish the proper strategy for the operator and the operation to be performed. This process is meant to last for ever. Every facility must elaborate its risks map and their communication to the operators based on the critical points of the installations and the measures adopted consequently as well as procedures fulfillments.

In this sense, the implementation of a survey that includes the necessary changes to avoid accidents or failures occurrence is necessary.

The strategy should include: [6] [7] [8]

- Debates on safety, leaflets, signals, warnings, etc.
- Demonstrations and evaluations based on incidents and/or situations that had occurred and lessons recently learned.
- Permanent supervision so as to detect and correct abnormal situations
- Control audits
- Procedures and intervention guides development
- Permanent training and retraining
- Procedures updating and revision

Table 1: Improvement proposals and its influence on the staff.

Negative aspects in the process of change	Positive aspects in the process of change
Uncertainty about the new process	Improvement of individual development
Unwillingness to add responsibility	Promotion of safety culture
Greater individual and group demands	Increase in training levels
Lack of knowledge about control systems (eg. type PLC)	Increase in the confidence on technology

This subject always arouses constant concern, especially about supervision, detection and correction of any deficiency. For this reason it is highly important to make evaluations of the operators performance, understanding and comprehension of risk that the task involves and the impact that will be generated if the duty is not performed with a responsible attitude towards himself.

5) Change attitude in the prevention of risks.

This task must be continuous. The attitude improves with development and the elaboration of strategies deeply related to prevention in facilities that use radioactive material for instance in a radiopharmaceutical production laboratory, in industrial radiography, etc. Uncontrolled situations should be carefully analyzed and informed among the staff so they can avoid any situations of the kind in the future.[1][2][10]

The models used to change the attitude should be simple so they can be easily implemented by the worker who is going to perform a task with radiation exposure risk. [11]

They should also reflect the politics that lead to improve the level of protection, influencing on the behavior of the operator and the assimilation of such politics on his part. (Table 2)

In case a dangerous behavior is detected it should be analyzed to determine the causes that caused it and whether the politics of prevention had been correctly set.

It is important to determine if the information, procedures and techniques that the installation possesses were clear enough for the operator and take the necessary corrective measures in order to avoid further major risk situations.

Finally, is very important that the information connected to the risk of using radioactive material and its consequences is personalized and preferently registered or documented.

Table 2. Benefits regarding the implementation of improvements in the operator's attitude.

Duties oriented to prevention	Results
Information	Updates and benefits individual and group development
Procedures	Avoids the possibility of radiological accidents for they establish a safe way to perform the task.
Risk evaluations	Determines the minimum training required for a given practice
Training	Allows evolution and constant improvement
Risk perception	Increases consciousness, attitude and respect for radioprotection principles.
Quality	Guarantees profitable results for the operator and the facility
Records	Allows task revision, failures detection and its analyses.

6) Human factor and accident rate.

Among the aspects that have to be taken into account in relation to this matter, we have to mention different situations that have to be avoided within the habitual work with radioactive material.

It is likely that to perform a certain practice all the procedures related have to be described, however, accident rates or frequency may not diminish in time. To believe that the operator is not conscious of risks or that he may not have interpreted the necessity to act towards radiological accidents prevention would probably be a simple criterion. This is when the so called human factor appears and it would be decisive to determine whether the decision taken was the correct one, according to the training the operator had received and the level of prevention established, being these aspects vital to determine whether they had avoided a probable accidental situation or contributed to make it happen. [11]

Regarding these aspects the following evaluations on the operator must be made:

- Psychological tests
- Degree of responsibility
- Personnel management
- Ability to analyze complex situations
- Ability in the communication of events towards personnel
- Responsibility in decision making

These studies will allow to establish whether the evaluated operator's profile is convenient to the capacity required for any given practice or process, determining whether his knowledge will allow him to fulfill the tasks with responsibility according to the risk the tasks implies.

7) Ethics of prevention.

During the last years an high degree of radiological safety was achieved by implementing changes oriented to deeper prevention politics where an increase in training as well as the development of general and specific procedures played the most important role.

Nowadays, many years after the risk of manipulating radioactive material was known it is necessary to get full knowledge and dedication to avoid incidental or accidental radiological situations. A proof of that is the great number of accidents regularly informed.

The opinion that the operator is the weakest link in the chain demonstrates the ethical compromise of acting consciously, respecting even the risks of the whole society and the environment. This belief demands a high commitment on the part of the operator but mainly the inclusion of mechanisms to analyze and guarantee that this condition will not be degraded in time. The ethic of prevention as a way of life leads the operator's attitude towards constant improvement as well as the compromise of the facility to avoid situations that generate risks to members of the public as the social responsible function incumbent on the facility. This will provide confidence and harmony in the development of any activity, independently of the social risk that it might generate.

8) Evaluation groups.

In every group in which radiological safety must be taken into consideration, the creation of evaluation committees is highly recommended. These groups should analyze different unexpected situations that might arise and which are not integrated or complemented in habitual procedures. The constitution of these committees must be the result of the analysis of the following aspects:

- Experience in decision making
- The criterion to solve the situation in a safe way
- The analysis of different proposals to achieve the same objective and its selection, getting the maximum benefit with the minimum effort.

Experience is the most important factor to take into account from the radiological safety point of view when solving a complex situation. Similar situations with satisfactory results are likely to be suggested once the benefit they had brought up is analyzed. The most interesting situations are those which provide new information or without previous knowledge. In this regard another important factor comes up and it is the capacity to judge. This capacity to judge the situation will modify the development of the action taken where the most important hint will be the resolution of the situation given with the lowest risk for the worker occupationally exposed and the installation. It is important to highlight that the decision to solve the situation or incident will have to be the most beneficial even when it is not the most economical. For instance, important contamination in a site due to an operator's mistake or duty using radioactive material out of a human error or any system o mechanism and its permanent solution. [11]

9) Preventive, predictive and corrective maintenance.

Maintenance duties during the life time of a facility, for instance a nuclear power plant, those places where industrial measuring instruments are installed are the ones representing higher risks for the operator because he will be in contact with structures, pieces or mechanisms that deteriorate out of their use, corrosion or material fatigue, which can be contaminated or generate important radiation levels. It can even be necessary to remove those pieces or mechanisms. This is the reason why the concept of design has to be high enough to assure a commitment to avoid undesirable doses and an adequate level of maintenance tasks to reduce possible risks.

It is important to highlight that general and particular audits are needed. Those aspects that need certain schedule and frequency are going to be audited during general inspections.

During the revisions, equipment and systems control is made in a thorough way by internal or external intervention groups that audit the quality of the processes in previous maintenance tasks.

Regarding these subjects, training of the personnel performing the tasks is relevant, as well as the profile of the operator that supervises the duties and frequency of maintenance tasks.

Additionally, priorities and protection rules to safeguard operators will be taken into account.

9.1) Preventive maintenance.

Preventive maintenance is based on techniques and processes that allow the increase of the life of a given component till the moment that failure which do not represent important risks for the operator and the facility become evident and its replacement turns necessary.

Preventive maintenance is based on the idea that by means of periodical inspections certain failures that lead the staff to take measures in the short term should be avoided. It also includes permanent evaluation criteria and periodic audits this maintenance must take into account the replacement of components or systems that guarantee that the replacement made allows the equipment or original system functioning. In the same way, the degree of confidence of the replacements and particularly of those which functions are very important for radiological purposes must be evaluated and registered so as to have backgrounds that allow the early detection of failures and the prediction of their behavior in time.

9.2) Predictive maintenance.

This kind of maintenance is closely related to the whole knowledge of a device, equipment or installation. In some cases it can be considered a part of preventive maintenance and it is mainly used when some indicators of degradation appear so it is possible to make the prediction before the malfunction occurs allowing to elaborate logic of maintenance.

This evaluation includes a great quantity of studies or diagnosis techniques and analysis, allowing the extension of the life time of an installation, with programmed intervention, which in most cases reduces the exposition of workers in a radioactive installation or one that uses processes that involve radiological risks out of unforeseen situations or urgency necessities, generating a concept of prevention to avoid unnecessary doses for the exposed worker during maintenance duties.

This activity is intimately related to ALARA concept for the worker and the inclusion of these concepts results indispensable.

9.3) Corrective maintenance.

This is the kind of maintenance made once the failure took place and it is impossible that the facility goes on operating. It is important to remember that there must exist clear particular procedures and written work authorizations to “work lonely” or far from a supervisor’s or responsible for the task that involves radiological risk, in such works when the place conditions require additional care and/or important risks. Once the replacement of the equipment or piece which must guarantee an identical performance to the faulty one, several tests have to be made (temperature, pressure, etc.) and the compliance of all parameters and necessary controls to approve the substituted safety mechanism and verify all safety requisites as well as the performance so as to confirm they are equivalent to the original one.

10) ALARA principle. (As Low As Reasonably Achievable)

ALARA principle introduces the concept of minimizing the biological risk of radiation exposure as much as possible bringing in reasonable measures that protect workers.

This concept is closely related to radiological safety principles and those measures that tend to guarantee the lowest risk for the best possible way to protect. [12][3][6][7]

By applying this concept several basic measures turn up:

- Reduction in time of exposure to radiation
- Reach to higher level of training
- Resort to training and practices that offer the minor risk for the worker involved so as to diminish the received doses.
- Use proper shieldings, according to the type of radiation involved in the practice.
- It includes the application of design, construction, operation and maintenance protection concepts of the facilities using radioactive material.
- Prevention from unnecessary or unjustified exposures.
- To communicate the involved staff exposed to radiation about the radiological risks and the measures or procedures applied in these cases.
- Provide a feedback mechanism regarding experiences related to ALARA

- Evaluate the attitude of the worker towards self protection and the protection of those cooperating in an efficient way.
- Generate a permanent evolution of procedures

These principles must be applied to reduce to the minimum workers risk to radiation exposure.

Conclusions.

Every remark made up to now are applicable to any industry, technology, process or practice using radioactive material at present.

Among the most important the following deserve to be highlighted:

- To make a risk evaluation at the moment of designing a practice or task involving the use of radioactive material, establishing risk maps to determine and define those tasks with dose compromise demand.
- Use models to predict a probable event that allows to evaluate possible radiological consequences.
- Establish as a priority to work towards workers, members of the public and the environment protection.
- To make operators conscious through safety debates, leaflets, instructions and permanently supervising his duty and correcting any deviation from safety procedures.
- Permanent training and re-training analyzing operator's aptitude towards radiological safety, his qualification and suitability.
- Write clear procedures and understandable for the operator, making verifications or audits to check their fulfillment.
- Detect and analyze possible failures in all safety systems used guaranteeing at least the substitution or replacement of the faulty ones by other similar components with the same level of confidence and safety.

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Medical Response in Radiation Emergency in Argentina

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Medical Response in Radiation Emergency in Argentina

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Abstract 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. ARN has a system for intervention in radiological -and nuclear emergencies with a primary intervention group, which is on duty in weekly shifts all year round. This paper aims at describing the system as implemented at present.

The Emergency Medical System has been developed into three levels:

Level I: local emergency services. This level includes triage (conventional and radiological), first-aid care, and first management of contaminated victims

Level II: emergency departments of local general hospitals that are in charge of performing a second triage by a biomedical approach, the treatment of conventional and/or radiocombined injuries and completing decontamination as necessary. In this way the initial triage is completed by a physical examination, timing and severity of prodromal signs and symptoms, sequential blood counts and serum enzymatic levels that allow a first-stage dosimetric approach at this level. Victims requiring higher complexity assistance shall be transferred to third-level hospitals.

Level III: three central reference hospitals (Hospital Naval "Pedro Mallo", Hospital de Quemados from Gobierno Autónomo de la Ciudad de Buenos Aires and Hospital Británico de Ciudad de Buenos Aires) capable of providing healthcare for diagnosis and treatment of acute radiation syndrome, cutaneous radiation syndrome and internal contamination constitute this level.

An educational program for medical and paramedical responders is regularly carried out at the three levels, including theoretical background as well as practical training. Guidelines and protocols for medical handling of victims have been drawn up. Research and development of new strategies for first medical response, diagnosis and treatment of radiation injuries are promoted by ARN in close collaboration with physicians belonging to third-level hospitals.

KEYWORDS: *radiation accidents, medical response*

Introduction

Although radiation accidents are not frequent, the increasing use of radioisotopes in medicine, research, and industry, as well as the growing reliance on nuclear power, increases the likelihood of these situations [1]. Additionally, risks posed by the use of radioactive sources for malevolent purposes have been highlighted during the last few years. In this context, the risks associated with radioactive sources have been the subject of increasing attention during the last decade and there are diverse scenarios that could lead to medical radiation emergency.

Experience has demonstrated a need for planning medical response, for this reason, the enhancement of national capabilities for medical assistance of casualties in radiation or nuclear emergencies becomes relevant. In our country the medical response in these situations has been organized in compliance with national capabilities for medical assistance. This communication presents the organization of medical response in nuclear and radiological emergencies developed in Argentina.

Discussion

Medical response to radiation accidents in Argentina has been organized according to The Nuclear Federal Law N° 24804. In this law 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: Radiological Emergency Intervention System/Nuclear Emergency Intervention System (SIER/SIEN). Both have primary intervention groups which are on duty in weekly shifts all year round. As part of this intervention system and as scientific and technical support the ARN/Radiopathology laboratory is in charge of:

- coordinating medical response in radiation emergencies
- making arrangements for providing specific supplies and equipment
- performing research programs concerning diagnostic and therapeutic options
- improving professional expertise on Radiopathology
- elaborating recommendations, guidelines and protocols for diagnosis and treatment of radiation injuries
- promoting education and training programs for personnel's healthcare involved in radiation emergencies

Other laboratories belonging to the ARN are in charge of dose reconstruction by physical dosimetry and biological/cytogenetic dosimetry, evaluation of internal contamination and bioassays.

The national system for medical response in radiation emergencies has been organized at four levels:

1.-First level response-Pre hospital response

Pre-hospital response may be given on-site by the relevant facility medical service or by local emergency medical systems. In Argentina the first-aid response includes physicians as part of the team. At the first level a conventional triage is performed with first-aid assistance of the life-threatening injuries, followed by a radiological triage, implementation of initial decontamination and transportation of casualties, when needed, to the hospital emergency department.

Training activities are regularly organized with the systems for emergency medical assistance, as a result of the cooperation promoted by ARN. Many guidelines for on-site management and transportation of radiation victims have been included as protocols or procedures [2, 3]

2. - Second level-Medical response at local general hospital

Local hospitals emergency departments are in charge of doing the second conventional/radiological triage, including treatment of conventional and/or radiocombined injuries, completing external decontamination as necessary and evaluating and treating of prodromal symptoms. Reception in an appropriate area for radiation victims is performed, taking into account radiation protection principles to avoid radioactive contamination spread [4]. The initial triage is completed by a biomedical approach through 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 phosphates). Blood samples for cytogenetic dosimetry and HLA typing and other biological samples like nasal mucus, sputum, and urine may be collected at this level

As a result of these evaluations, there are three possible decisions for victims:

- may return home,
- may require hospitalization at the local general hospital,
- should be transferred to a high complexity hospital.

3. - Third level-Medical response at central high-complexity hospitals

Central high-complexity hospitals are capable of offering efficient healthcare to radiation victims with suitable infrastructure, equipment, human resources and professional expertise. In these hospitals, *ad hoc* committees of radiopathology composed of trained professionals with advice from ARN expertise make protocols for diagnosis and treatment of acute radiation syndrome, cutaneous radiation syndrome and internal radioactive contamination [5, 6].

Agreements on scientific and technical cooperation for Medical Assistance of Radiation Victims have been signed between the Nuclear Regulatory Authority and the following third-level

hospitals: Hospital Naval Pedro Mallo and Hospital de Quemados. These agreements include training of human resources and research activities on radiopathology.

Moreover, the ARN promotes the interaction between the Radiopathology and Biological Dosimetry Laboratories and other institutions with professional expertise concerning particular areas such as:

- Hematology: Bone Marrow Transplantation Units from hospitals in Buenos Aires
- Toxicology: Argentine Toxicology Network/REDARTOX
Toxicology Department of Juan Fernandez Hospital
Toxicology Cathedra of Buenos Aires University
- Pharmacology: School of Pharmacy and Biochemistry of Buenos Aires University
- Pediatric: Radiotherapy Department of the National Pediatric Hospital
- Psychological impact in emergency and disaster situations: Human Factors Group of the System for Emergency Medical Assistance/SAME
- Biological Dosimetry: ARN's biological dosimetry reference laboratory and, in process, the development of the technical competence of an associated laboratory IGEVET (La Plata National University-CONICET) for assistance when the capacity of the reference laboratory is exceeded. The ARN's biological dosimetry laboratory response is coordinated by ARN's Emergency Response System and works in cooperation with other regional and international assistance programs: Latin American network of Biological Dosimetry; RANET-IAEA; BioDoseNet-WHO

The response capability of the different areas is well established in order to allow the network healthcare services to select an appropriate cohort of individuals whose treatment may benefit from their expertise.

Education and training programs are regularly executed locally 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, exercises and emergency drills on Medical Response in Radiation Accidents. Modules concerning radiation protection, radiation biology and radiopathology have been included in the syllabus of postgraduate courses for burns and toxicology by 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.

4. - Fourth level- Regional and International Cooperation

Through the Regional and International Cooperation Argentina has taken part in:

1. The project RLA/9/031 "Medical Treatment in Cases of Radiation Accidents" was conducted 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[7]. 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. 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.
2. The ARCAL project RLA/9/045 "Enhancement and Harmonization of National Capabilities in Radiation Emergencies" was conducted 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.
3. As a Liaison Center for the Radiation Emergency Medical Preparedness and Assistance Network (REMPAN) coordinated by WHO's Radiation Program, the Radiopathology Laboratory together with Radiological/Nuclear Emergencies of the ARN take 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 [8], the Convention on Early Notification of a Nuclear Accident and the Convention on Assistance in the case of a Nuclear Accident or Radiological Emergency.

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 ARN in Argentina. As shown, medical response has been set up as a three-level system, taking into account the national needs and capabilities. Therefore instead of having medical facilities for the sole purpose of treating radiation injuries, this system has been based on adapting 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 preparedness and management in radiation emergencies.

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Argentine Experience on the Application of the Code of Conduct on the Safety and Security of Radioactive Sources and the Guidance on the Import and Export of Radioactive Sources

Vidal, D.N. and Massera, G.E.

ARGENTINE EXPERIENCE ON THE APPLICATION OF THE CODE OF CONDUCT ON THE SAFETY AND SECURITY OF RADIOACTIVE SOURCES AND THE GUIDANCE ON THE IMPORT AND EXPORT OF RADIOACTIVE SOURCES

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Abstract: Argentina has shown, historically, a great commitment with Safety and Security of Radioactive Sources, even before the international decision of developing and implementing a Code of Conduct, and the Guidance on the import and export of radioactive sources. In 1999 the Board of Governors of IAEA called for the development of a Code of Conduct on the Safety and Security of Radioactive Sources. The actual version of the Code of Conduct on the Safety and Security of Radioactive Sources was approved by the Board Governors of IAEA on 8th September 2003. It replaced the version published in March 2001. In 2004 the Board of Governors of IAEA approved the text of the Guidance on the Import and Export of Radioactive Sources. This document has been presented as a supplementary guidance to the Code of Conduct on the Safety and Security of Radioactive Sources. It is important to remark the big international support for the Code and the Guidance expressed for a great number of States, even though its application and implementation is not legally binding. During these years, after the approval of these documents the International Community has been acquiring a great experience on the application and implementation of the Code of Conduct and the Guidance. IAEA has proposed, during these years, to carry out international meetings in order that States can share experience in the implementation of both the Code and the Guidance. The International Community has answered to these proposals, in general, pleased and thankful because the opportunity. Some of these meetings have been Regional. The first intention of this paper is to: share the experience acquired by Argentina in this matter, through some examples; remark the benefits for the region to which Argentina belongs, in particular with those countries with which Argentina has commercial exchange, on the implementation of the Guidance on the Import and Export of Radioactive Sources; remark the importance of going on making efforts to encourage more States to act in accordance with the Guidance on a harmonized basis; remark some difficulties met in the application of the Guidance; and evidence some aspects on which it would be important to go on working with the objective of getting higher and higher safety and security of radioactive Sources levels. Is also in the scope of this paper to remark the importance of the assistance to the meetings and besides, especially in America's Community, improving communication between Regulatory Bodies on the Safety and Security conditions for the transfer of radioactive sources.

KEYWORDS: sources, export and import, code of conduct.

1. INTRODUCTION

Argentina has been engaged in trade activities of radioactive sources for many different applications since long time ago, becoming in particular one of the main world suppliers of Co-60 during the past two decades, trades always encompassed with a great commitment for Safety and Security of radioactive sources, even before the international decision of developing and implementing a Code of Conduct, and the Guidance on the import and export of radioactive sources.

The experience gain in dealing with safety and security of sources during import and export transfers was relevant not only for our contribution to the preparation of the Code of Conduct on the Safety and Security of Radioactive Sources and Guidance on the Import and Export of Radioactive Sources but also for encouraging its implementation.

This paper intends to show several aspects related to the implementation of the Guidance as was experienced by the Nuclear Regulatory Authority of Argentina, mainly to:

- share the experience acquired in this matter, through some examples.

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- remark the benefits for the region, in particular with those countries with which Argentina has commercial exchange, on the implementation of the Guidance.
- remark the importance of the efforts done to encourage more States to act in accordance with the Guidance on a harmonized basis.
- remark some difficulties met in the application of the Guidance.
- evidence of some aspects on which it would be important to go on working with the objective of improving safety and security of radioactive source levels.

Is also in the scope of this paper to remark the importance of the assistance to the meetings and besides, especially in America's Community, improving communication between Regulatory Bodies about Safety and Security conditions for the transfer of radioactive sources.

2- ARGENTINE EXPERIENCE IN THE USE OF EXPORT/IMPORT FORMS

After IAEA international meetings on aspects related to Import and Export of radioactive sources, they appeared, basically, three suggested forms to be used in the communication between exporting and importing countries, as follows:

- "REQUEST TO THE IMPORTING STATE FOR CONSENT TO IMPORT CATEGORY 1 RADIOACTIVE SOURCES"
- "REQUEST TO THE IMPORTING STATE FOR CONSENT TO IMPORT CATEGORY 1&2 SOURCES UNDER EXCEPTIONAL CIRCUMSTANCES"
- "NOTIFICATION TO THE IMPORTING STATE PRIOR TO SHIPMENT OF CATEGORY 1 OR 2 RADIOACTIVE SOURCES".

REQUEST TO THE IMPORTING STATE FOR CONSENT TO IMPORT CATEGORY 1 RADIOACTIVE SOURCES OR TO IMPORT CATEGORY 1&2 SOURCES UNDER EXCEPTIONAL CIRCUMSTANCES is a communication between Regulatory Authorities.

NOTIFICATION TO THE IMPORTING STATE PRIOR TO SHIPMENT OF CATEGORY 1 OR 2 RADIOACTIVE SOURCES is a communication that may be done by a Regulatory Authority or a facility to another Regulatory Authority or facility.

Argentina has adopted the agreed suggested forms as result of the IAEA meetings on the Guidance for Import and Export of radioactive sources. never the less it was felt the need to introduce some small changes to these forms in order to fulfill national requirements mostly related to information required to grant the final authorization.

Even though, Argentina had already in place, since long, written procedures to authorize import and export of radioactive sources it was necessary to make some formal arrangements to better approach the international commitments assumed. But in fact there was no need to introduce any significant change to the regulations in force.

It is important to remark that because of the independence of the Nuclear Regulatory Authority (it directly reports to the Executive) and through the mandate from the "Nuclear Activity Law" the Nuclear Regulatory Authority is entrusted to create its own regulations.

The fact that Regulatory Body in Argentina gathers all the activities related to the control of radioactive sources (assessment, licensing and control of radioactive sources) facilitates the proper implementation of the Guidance.

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Even though in recent years, many States have implemented the proposed forms, it is from our point of view, that the process has not yet reached the necessary maturity for a full international implementation. Several international meetings have been carried out in order to achieve this maturity. In these meetings, representatives from some hundred states have had the opportunity to share experiences on the matter not only during plenary sessions but during working sessions, in particular for the working group with representatives from the Latin American region. For us, the meetings have been of great help in order to realize the many different aspects that affect, sometimes notoriously, the import and export processes. For example, different cultures and ways of life; the relative geographical position of the country in respect of the routes of transport and different local issues related to the way organizations deal with the import/export processes.

These meetings have joined big export countries with smaller ones and with countries for which export is not a commercial activity. In the case of the Spanish language meetings, even though very few people assisted it, allowed us, to get a closer relationship with the representatives of countries with which Argentina has an important trade that requires a substantial import/export control activity. In order to reach full implementation of the consent procedure it is necessary that all the states become fully acquainted with the meaning of the measures, that is to understand the exact interpretation of import consents and be able to share these interpretation with the other states.

In our experience, it was of great importance networking through which we could achieve a closer and more reliable communication supported at least in the first instance by the Contact Points.

Related to the export or import of Category 1 radioactive sources currently most of the involved export countries for this category of sources are applying the suggested forms, but this was found not to be the case for countries with very small amount of source transfer.

In some cases, our experience has shown that the role of request for the consent previous to any import/export authorization has been assumed by the facility rather than the National Authority while in other cases the “Notification to the importing State prior to Shipment” form was used instead of the corresponding prior request for consent.

In our experience, as an export country, it has happened that:

- requests for consent have been granted directly through an Import Authorization issued by the National Authority of the Import State;
- there has been no reply to the request for consent or;
- the Regulatory Authority of the importing country, rejects using the suggested forms for the consent, instead, they simply reply by stating; “the State has no objections on the import of such radioactive sources”.

Regarding our experience the concept of the request for consent between states has grown up but its implementation is not uniform depending on the region and even mixed with what it seems to be great cultural influence.

3. EXPERIENCES IN THE USE OF GUIDANCE ON IMPORT/EXPORT OF RADIOACTIVE SOURCES

The Nuclear Regulatory Authority as a petitioner of the consent is engaged in the control of radiological safety aspects that facilities which intend to export radioactive material must accomplish.

In Argentina, when dealing with the import of radiation sources, the objects of analysis prior to granting consent are: a) the institution that is going to import radioactive material, which may be or not the same that is going to use it, b) the facility that will use it, when the final consignee is known,

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and c) the national laws and regulations. The aspects covered are those related to the radiological safety and security of the facility. The most important is to verify compliance of them, holding the necessary licenses in force.

There may be cases in which consent for import cannot be granted due to the fact that the intended recipient party does not meet regulatory requirements and is temporarily lacking of, for example, an authorization or permit. This situation about the importer is hard to be distinguished by the consignor. The operation license may be suspended because of unacceptable radiological safety circumstances that were noticed in opportunity of an audit. This includes, for instance, that the radiological safety responsible individual permit be expired.

Any of these situations may occur at any time, but the analysis at the moment of the corresponding request for consent to import radioactive material is an opportunity that the import State has to prevent the occurrence of a complex circumstance, probably difficult to solve from the radiological safety point of view in the case that the material be sent without prior notice.

In our opinion this is an advantage when a request for consent is done on a fixed export by export basis schedule.

It also seems more convenient that the request for consent includes the probable date of export/import. A great added value must be recognized to the consent process in that to reply a request for consent it implies verification of compliance with regulations by all those involved not only in the import process but also for the use and disposal, if the case, of the source.

This verification process is also in line with mandatory requirements for domestic transactions in Argentina, where there is also a prohibition to transfer any radioactive material or source to an installation without an operation license in force. This implies notification that allows keeping track of the material involved.

In this sense, the most important sellers regarding amount of radioactive materials “consumable” count on informatics systems that only allow providing them under those conditions. Suspensions of Operation licenses are informed by the Regulatory Authority directly to the sellers, so they can suspend the supply.

There are other certain cases in which the seller of radioactive material or of the equipment containing radioactive material requires from the institution that demands the material an authorization by the Regulatory Authority, in order not to exceed the maximum content of radioactive material of the License of Operation.

These are different variants, equivalents to the request for consent /consent, between countries, very easy to use within the country but more complex when the exchange takes part between different States.

However, it is important to analyze the domestic procedures for, in some cases, they can be valid alternatives to the request for consent system between countries with a high degree of integration, whose regulatory authorities, besides, share a similar licensing system, rules, sanction regime and shared procedures.

Argentina has received, with great interest, other countries requests for bilateral agreements that facilitate the export/import procedures without setting back the existing commercializing mechanisms but assuring safety and security conditions and in this direction intends to work. Argentina is working on the implementation of common procedures in order to facilitate the import/export processes mainly with countries with which a relevant amount of trade is currently held.

Given the recognized relevance of the request for consent process in radiation safety it is in this sense though that would be worth to extend the scope of the Guidance on import/export of Categories 1 and

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2 radioactive sources when these must be transported through other States than from the Import/Export engaged ones.

Argentina has been, in case of Category 1 sources, a transit country.

In particular for the case of Category 1 sources, our experience in the matter shows it would be highly convenient in order to ensure adequate regulatory control, that the transit country be involved in the process of request for consent, to ensure verification of proper compliance safety and security measures. Even though the notification of transit countries is included in the International Transport Regulations they do not cover all the Category 1 or 2 sources.

In Argentine, we had the case of an import shipment of category 1 radioactive sources in which the preparedness for the consignment procedure, implemented by the consignor, was not appropriate. In this case, the communication with the point of contact of the export State was useful in order to award the export regulatory body about the non compliance in order to avoid the occurrence of situations like this in future source shipments.

The fact that States make the communications through their contact points make it easy to initiate conversations between Regulatory Authorities so first they can explicit the problem and second, after the analysis and understanding of it, the Regulatory Authority of the export country can take actions, such as making the follow up to avoid the occurrence of these situations.

Guidance is focused up to now, on the import/export process, but does not give any guidance on the way to close this process. The import/export finishes when the radioactive sources arrive in safe and secure conditions to its destination. Even though it is expected that the consent procedure would be applied between states for the safe transfer of sources which includes notification prior to shipment, there is no recommendations about further communications confirming the arrival to the final user. This requisite that would provide a verified end to transfer process is a lacking issue in the Guidance on import/export of radioactive sources that should be included in future revision of the document.

4. OTHER EXPERIENCES

In Argentina Licensees are entrusted with the responsibility and commitment on the full compliance with safety requirements applied on destination of any not exempted radioactive material that is being transferred. Meaning that for those cases in which the radioactive material is neither Category 1 nor Category 2 the processes of authorization of export on the part of the Regulatory Authority becomes involved in collecting and providing information through the contact points from import States in order to diminish the probability of radiological safety or security breaches incidents.

This has helped to avoid the export of radioactive material to facilities that did not hold an operation license or even worst when trying to apply for authorization with fake documentation.

In many cases this exchange of information between contact points has allowed to improve the information provided in the documents required for granting import authorizations as well our control process as export state.

Import authorizations, granted by the regulatory authority of the import State presented by the export licensee, to the regulatory body in Argentina may contain a variable combination of data about the radioactive material that is authorizing: (p.i.: radionuclide description as physical form, maximum allowed activities; dead line for the import license; identification of export facility; identification of import facility; identification of a responsible officer from the import installation).

Due to this situation we had to revise export authorizations procedures mainly for transfers to neighbor countries, in the region, taking into account the characteristics of the authorizations granted by their regulatory bodies in order to assure that the export licensee is consigning radioactive material that the recipient is allowed to receive.

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So far and given that the consent method being implemented is still in an early face we have taken a flexible attitude regarding administrative control of import/operation licenses (awarded by import countries) presented by the export licensee, never the less stressing the control in all the stages of the transfer process for which and in any degree of doubt, the contact point of the import country is consulted for an adequate and clear interpretation of the information made available in the documentation provided.

5- CONCLUSIONS AND RECOMMENDATIONS

In our experience there is not, up to now, an absolute and common interpretation between parties on the relevance that the request for consent implies, that is, the scope and consequences implied as well as which organizations are expected to formulate the request for consent and to give the respective consent.

There is a need to stress on the importance of reaching an international commitment with the implementation of import and export procedures through a full common interpretation of its terms, scope and usefulness.

It would be appropriate to discuss and, if necessary, to update the Guidance on the import/export of radioactive sources taking into account the transit countries and the final notification of the radioactive sources including any aspect to be taken into account on the safety and security conditions on arrival. Multilateral agreements are considered functional for a better understanding of the terms of the Guidance and should be promoted as an excellent complement of the international meetings.

Request for consent on an export by export basis has proven to be a important tool for improvement of regulatory activities in particular when, within the organization, different aspects of concern to grant import/export authorization such as compliance with safety, security and transport regulation need to be coordinated.

Points of contact are of great value in the import/export process, even though they refer to Categories, other than 1 and 2.

The exchange of information between contact points gives opportunities to improve the Argentine control process as export State as well the data contained in the documents granted for the import State authorizations.

Finally the need to proceed with the contact points for granting consent on import/export is a fact that by bringing closer communication between responsible parties is contributing to reach a global understanding for an effective control of radioactive sources.

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Adaptation of Regulatory Information and Knowledge through Knowledge Maps in the Argentine Nuclear Regulatory Authority within the Framework of Nuclear Renaissance

Chahab, M. and Dawyd, N.

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El presente trabajo presentó poster durante el evento.

Adaptation of Regulatory Information and Knowledge through Knowledge Maps in the Argentine Nuclear Regulatory Authority within the Framework of Nuclear Renaissance

Chahab, M.¹ and Dawyd, N.

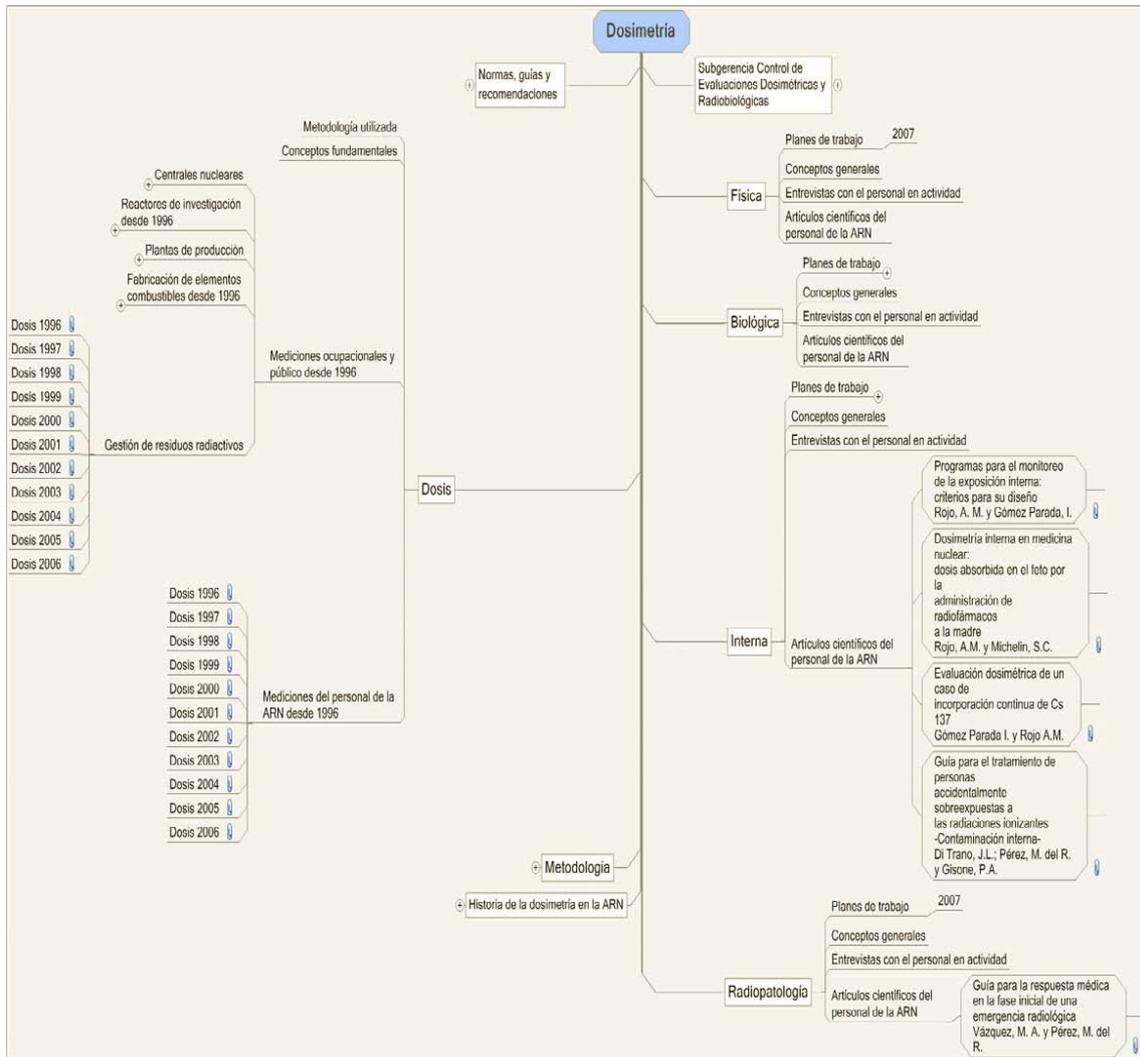
Nuclear Regulatory Authority
Argentina

Abstract

In the new framework of nuclear renaissance in the world in general, and in Argentina in particular, proper and efficient management of information and knowledge produced in the past and to be produced during renaissance becomes critically important. The fact that in the nuclear sector across the world human resources are going through significant change as a result of the massive number of experts who are retiring from the workforce, the ensuing general gap, the new generation of workers who are joining the nuclear rank and file with different training, values and cultural beliefs, and the slow information and knowledge transfer process call for carefully considering and assessing new methods to manage information and knowledge. This paper discusses the topic of knowledge maps as a method to adapt historical information and knowledge and to make it more readily available for future workers; the paper also deals with a new management approach to such information. Knowledge maps probably represent an up-to-date method to manage both historical and new information and knowledge, adapting to a number of new cultural features, including but not limited to the intensive use of information technologies and the tendency to summarize and integrate concepts. A distinguishing feature of this new method of organizing information and knowledge is the need for a closer interrelation across the organization's sectors. As a result, knowledge maps help create and improve manuals and procedures related to the specific tasks performed in the institution, based on the analysis carried out by those creating the maps. This tool also helps better analyze the tasks already conducted or to be conducted by workers, all of which optimizes the job description process in the area of human resources. Another benefit of knowledge maps is that they help preserve the information and knowledge that can be used to train the staff in merely technical or induction issues as well as in an almost immediate search for information for organizational decision-making purposes. Finally, knowledge maps contribute new specialist-centered or subject-specific information to libraries. Knowledge maps show a universe of information and knowledge in a summarized and orderly manner, helping the institution work more effectively and efficiently. As an example of the proposition, a map of one of the topics carried out at the Argentine Nuclear Regulatory Authority is illustrated.

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An example of knowledge map:



Adaptation of Regulatory Information and Knowledge through Knowledge Maps in the Argentine Nuclear Regulatory Authority within the Framework of Nuclear Renaissance.

12th International Congress of the International Radiation Protection Association



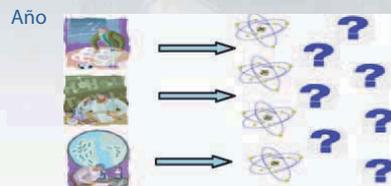
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Introduction

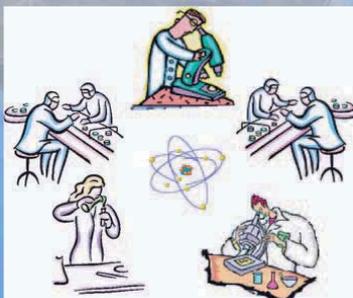
In the new framework of nuclear renaissance in the world in general, and in Argentina in particular, proper and efficient management of information and knowledge produced in the past and to be produced during renaissance becomes critically important. The fact that in the nuclear sector across the world human resources are going through significant change as a result of the massive number of experts who are retiring from the workforce, the ensuing general gap, the new generation of workers who are joining the nuclear rank and file with different training, values and cultural beliefs, and the slow information and knowledge transfer process call for carefully considering and assessing new methods to manage information and knowledge.



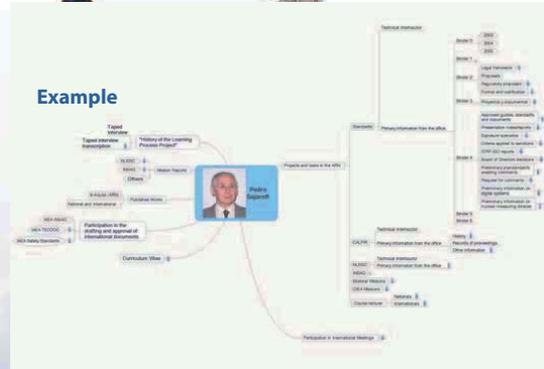
Methodology

The topic of knowledge maps as a method to adapt historical information and knowledge and to make it more readily available for future workers. Knowledge maps probably represent an up-to-date method to manage both historical and new information and knowledge, adapting to a number of new cultural features, including but not limited to the intensive use of information technologies and the tendency to summarize and integrate concepts. A distinguishing feature of this new method of organizing information and knowledge is the need for a closer interrelation across the organization's sectors.

Año



Example



Benefits and Results

As a result, knowledge maps help create and improve manuals and procedures related to the specific tasks performed in the institution, based on the analysis carried out by those creating the maps. This tool also helps better analyze the tasks already conducted or to be conducted by workers, all of which optimizes the job description process in the area of human resources. Another benefit of knowledge maps is that they help preserve the information and knowledge that can be used to train the staff in merely technical or induction issues as well as in an almost immediate search for information for organizational decision-making purposes. Finally, knowledge maps contribute new specialist-centered or subject-specific information to libraries. Knowledge maps show a universe of information and knowledge in a summarized and orderly manner, helping the institution work more effectively and efficiently.



Radiological Assessment of a PET Facility

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Radiological Assessment of a PET facility

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Abstract. The radiological assessment of a Positron Emission Tomography (PET) facility consists of the evaluation of the annual effective dose to workers exposed occupationally and to members of the public. This evaluation takes into account the radionuclide involved, the characteristics of the facility, the working procedure and the expected number of patients per year.

This paper details the methodology used by the Nuclear Regulatory Authority (in Spanish ARN) to independently assess the design of PET facilities considering only radioprotection aspects. The results of the evaluation are compared with the design requirements established in the ARN regulations to determine whether or not, the facility complies with those requirements, both for workers and for members of the public.

As an example of the above mentioned methodology, this paper presents the assessment of a PET facility located in Buenos Aires called *Fundación Centro Diagnóstico Nuclear* (FCDN).

KEYWORDS: ¹⁸F; PET; shielding; nuclear medicine

1. Introduction

It is one of the functions of the ARN to regulate and to control the nuclear activities concerning radiation protection issues. For this purpose, the ARN develops regulatory systems, which are applied to all the nuclear activities carried out in Argentina. Nuclear medicine facilities are found in ARN's regulated field, in particular PET centers.

The PET diagnostic technique is based on the detection of photons annihilation of a positron emitter previously administered to a patient. The positron-emitting radionuclides used have typically short half-lives, such as ¹⁵O, ¹³N, ¹¹C and ¹⁸F, which is the most versatile clinical PET radiopharmaceutical. Because of its relatively long half-life compared to the other commonly used positron-emitting radionuclides, as it can be seen in Table 1, the expected radiation protection measures, assuming that ¹⁸F is always used, should be adequate for studies where the same activity of shorter-lived radionuclides are applied. This work focuses on PET centers that use ¹⁸F.

The radiological assessment consists of the evaluation of the annual effective dose to workers exposed occupationally and to members of the public. This evaluation takes into account the radionuclide involved, the characteristics and lay-out of the facilities, the working procedures and the expected number of patients per year. The evaluation embraces the distributions of rooms, the thickness and physical material of walls, floors and ceilings. The working procedures give information of ¹⁸F activity administered per patient; the duration of time that each injected patient remains in the waiting room (uptake time), the duration of time of image acquisition (acquisition time), and so on. Occupational factors of the different areas are considered in the evaluation.

The Basic Safety Radiation Standard AR 10.1.1 establishes that radiation protection shall be optimized and define dose constraints for members of the public and workers exposed occupationally. However article 89 gives a design option that may be used as an alternative to demonstrate that the radiation protection system are optimized. The text in article 89 may be translate as follows:

“When the design of the protection radiation systems ensures that, under normal operation conditions, no worker will incurs in an effective dose higher than 5 mSv in a

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year, no member of the public will incur in an effective dose higher than 100 μSv in a year and that the collective effective dose per year of operation is lower than 10 man-Sv, it is not necessary to prove that the systems are optimized unless the ARN required otherwise.”

At most designer uses this option, the results of the evaluation are compared with these design dose constraints to determine whether or not, the facility complies with those requirements, both for workers and for members of the public. Due to the characteristics of PET studies, the sources that are taken into account are: 1) the patient at rest (uptake time) and 2) the patient in PET scanner (acquisition time).

2. Evaluation procedure

2.1 Parameters

The evaluation procedure considers the following parameters:

2.1.1 Effective dose rate constant, Γ

The effective dose rate constant of a source is the effective dose per unit of time and per unit of activity at 1m from the source in case of an antero-posterior irradiation geometry (A-P). The effective dose rate constant for each radionuclide used is calculated according to the conversion factors provided by ICRP Publication 74 [3].

2.1.2 Administered Activity, A_0 / Uptake time, t_{up} / Acquisition time, t_{ac}

The amount of activity administered to the patient depends on several factors such as mass patient, mode imaging, PET scanner, and so on. Generally adults receive 370-740 MBq (10-20 mCi) of ^{18}F while children receive 4-5 MBq / kg. At the moment of the image acquisition the activity on the patient is

$$A(t_{up}) = A_0 \cdot e^{-\lambda \cdot t_{up}} - \alpha \cdot A_0 \quad (1)$$

where $A(t_{up})$ is the remaining activity in the patient at the uptake time t_{up} ; λ is the decay constant; A_0 is the administered activity and α is the fraction of the administered activity excreted by the patient before the acquisition of the images (typically $\alpha=0.15$).

The uptake time usually is 60 min and acquisition time varies between 20 and 30 min. The information of these three parameters (the administered activity, the uptake time and the acquisition time) should be provided by the facility.

2.1.3 Transmission factor, k

The transmission factor is defined as

$$k = \frac{D(r_o)}{D_o(r_o)} \quad (2)$$

where $D(r_o)$ is the dose at an interest point considering the shielding and $D_o(r_o)$ is the dose in the same point without the shielding. In the evaluation of a PET facility, the building shields are considered as well as the patient attenuation (self-attenuation). Typical materials used as shielding in a PET facility are lead, iron, concrete, and masonry (common brick). The k factors applied were calculated by the Monte Carlo method [2] considering a broad beam geometry. The self-attenuation depends on the energy of the radiation source inside the patient. Studies found in the literature [4] indicate that the transmission factor of a typical patient is approximately $k_{pat} = 0.64$ for radionuclides used in PET with

a characteristic energy of 511 keV For masonry, the transmission factors of concrete, modified by the relationship between the densities, are used^[5].

$$\frac{\delta_{mamp}}{\delta_{concr}} = \frac{X_{concr}}{X_{mamp}} \quad (3)$$

where δ_{mamp} is the masonry density; δ_{concr} is the concrete density; X_{mamp} is the thickness of masonry and X_{concr} is the thickness of concrete. In case of multilayer shields, the total transmission factor is obtained as a product of the individual transmission factors (included k_{pat}).

$$k_{tot} = \prod k_i \quad (4)$$

where k_{tot} is the total transmission factor for every shield and k_i are the transmission factors of the individual ones.

Due to the high penetration of gamma radiation involved in the PET process, it is necessary to contemplate the rooms at the same level and also those located above and below it.

2.1.4 Dose reduction factor, R_t

Since the radionuclides used have short half-life, it is necessary to consider their decay when the dose is integrated. For this purpose the dose reduction factor is defined as follows

$$R_t = \frac{E(t)}{\dot{E}_0 \cdot t} = \frac{1}{\lambda \cdot t} \cdot (1 - e^{-\lambda t}) \quad (5)$$

where $E(t)$ is the effective dose accumulated during a time t ; \dot{E}_0 is the initial effective dose rate and λ is the decay constant.

2.1.5 Occupancy factor, T

The occupancy factor for an area is defined as the fraction of effective irradiation time while an individual remains in the area or point of interest. The occupancy factor depends on the facility itself and the work procedure. In that cases where this information is not available it is possible to apply the T suggested in the bibliography^[5].

Table 1: Positron emitters radionuclides used in PET studies.

Half life ($T_{1/2}$); emissivity (ϵ); dose reduction factor (R_t) and effective dose rate constant (Γ).

Radionuclide	$T_{1/2}$ [2]	Photon energy [2] [MeV]	ϵ [2]	Γ^a [mSv.m ² /h.MBq]
¹¹ C	20.4 min	0.511	2.00	1.44E-04
¹³ N	10.0 min	0.511	2.00	1.44E-04
¹⁵ O	2.0 min	0.511	2.00	1.44E-04
¹⁸ F	109.8 min	0.511	1.93	1.39E-04
⁶⁴ Cu	12.7 h	0.511 / 1.346	0.38 / 0.005	2.70E-05
⁶⁸ Ga	68.3 min	0.511	1.84	1.33E-04
⁸² Rb	76 s	0.511 / 0.776	1.90 / 0.13	1.50E-04
¹²⁴ I	4.2 d	0.511 / 0.603 / 1.693	0.5 / 0.62 / 0.3	1.45E-04

^a ICRP 74 in A-P geometry

2.2 Assessment methodology

2.2.1 Model

The annual effective dose at a point of interest due to an operation with a given source is calculated as follows

$$E = \Gamma \cdot \frac{A \cdot t_{op} \cdot N}{d^2} \cdot k \cdot R_t \cdot T \quad (6)$$

where Γ is the effective dose rate constant of the source; A is the administered activity of the patient; t_{op} is the time of the operation; N is the number of operations performed annually; d is the distance between the source and the point of interest; k is the total transmission factor taking into account all shields between the source and the point of interest; R_t is the dose reduction factor and T is the occupancy factor at the point of interest.

The operations considered are: 1) the patient at rest (uptake time) and 2) the patient in PET scanner (acquisition time). The annual effective dose is calculated by adding the doses caused in both situations.

2.2.2 Assessment

The annual effective dose results are compared with the design dose constraints required by the standard AR10.1.1. These dose constraints are:

- Annual effective dose for workers exposed occupationally: 5 mSv
- Annual effective dose for members of the public: $100 \text{ } \mu\text{Sv}$
- Annual collective effective dose: 10 man-Sv

It can be shown (for point sources or shielding rooms) that, if a facility complies with the first two restrictions, consequently it meets the third.

3. An example: *Fundación Centro Diagnóstico Nuclear*

This paper presents the assessment carried out in *FCDN*. This PET facility is located in Buenos Aires and it has a Positron Emission Tomograph together with a Helical Computed Tomograph (PET-CT). The *FCDN* receives 500 mCi of ^{18}F , 200 mCi of ^{13}N and 100 mCi of ^{11}C three times per week. Firstly the calculation is done considering 800 mCi of ^{18}F , according to what was stated in point 1. Because of its relatively long half-life, compared to the other commonly used positron-emitting radionuclides, the expected radiation protection measures, assuming that ^{18}F is always used, would be adequate for studies when the same activity of shorter-lived radionuclides are applied. The main characteristics of the facility are^[6]: 1200 patients per year; 555 MBq (15 mCi) of administered activity; 60 min of uptake time and a 30 min of acquisition time. Figure 4 shows the room layout of the facility and the points of interest. Tables 2 and 3 show a summary of the calculations of annual effective dose to the public and workers.

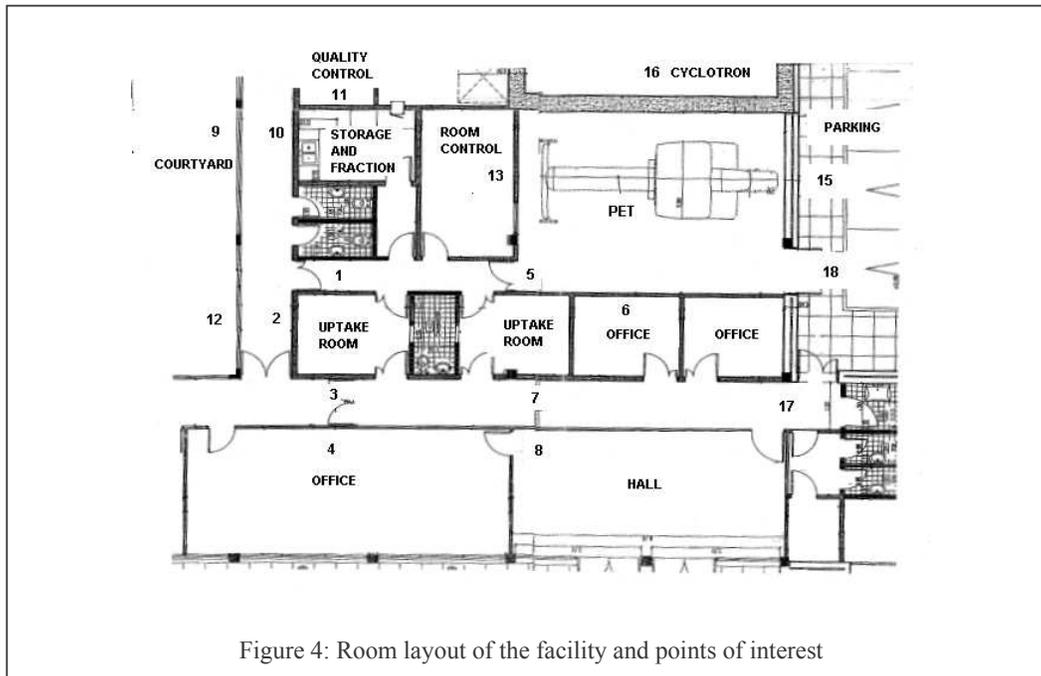


Figure 4: Room layout of the facility and points of interest

Table 2: Effective annual dose for workers

Administered activity (A); thickness of lead (X_{pb}); equivalent thickness of concrete (X_C); lead transmission factor (K_{pb}); concrete transmission factor (K_C); patient transmission factor (K_{pat}); dose reduction factor (R_t); occupancy factor (T); distance source-interest point (D); operation time (t_{op}); annual number of patients (N); effective annual dose (E).

LOCATION	OPERACIÓN	A [mCi]	X_{pb} [cm]	X_C [cm]	K_{pb}	K_C	K_{pat}	R_t	T	D [m]	t_{op} [min]	N	E [mSv]	E tot [mSv]
1 Corridor	Uptake (L)	15	1	6	2.48E-01	6.52E-01	0.64	0.832	1/5	0.8	60	600	1.25E+00	1.38
	Uptake (R)	15	0	0	1	1	0.64	0.832	1/5	6.5	60	600	1.17E-01	
	Acquisition	8.7	0.3	6	6.90E-01	6.52E-01	0.64	0.911	1/5	10	30	1200	1.41E-02	
2 Corridor	Uptake (L)	15	1	10	2.48E-01	3.98E-01	0.64	0.832	1/5	0.8	60	600	7.60E-01	0.78
	Uptake (R)	15	1	10	2.48E-01	3.98E-01	0.64	0.832	1/5	7	60	600	9.93E-03	
	Acquisition	8.7	0.3	10	6.90E-01	3.98E-01	0.64	0.911	1/5	10	30	1200	8.61E-03	
3 Corridor	Uptake (L)	15	3	10	1.14E-02	3.98E-01	0.64	0.832	1/5	1.5	60	600	9.94E-03	0.15
	Uptake (R)	15	0	0	1	1	0.64	0.832	1/5	6	60	600	1.37E-01	
	Acquisition	8.7	0.3	10	6.90E-01	3.98E-01	0.64	0.911	1/5	11	30	1200	7.12E-03	
5 Corridor	Uptake (L)	15	0	6	1	6.52E-01	0.64	0.832	1/5	6.5	60	600	7.61E-02	1.52
	Uptake (R)	15	1	6	2.48E-01	6.52E-01	0.64	0.832	1/5	0.8	60	600	1.25E+00	
	Acquisition	8.7	0	0	1	1	0.64	0.911	1/5	4	30	1200	1.96E-01	
6 Office	Uptake (L)	15	1	6	2.48E-01	6.52E-01	0.64	0.832	1	8	60	600	6.23E-02	2.86
	Uptake (R)	15	1	6	2.48E-01	6.52E-01	0.64	0.832	1	1.3	60	600	2.36E+00	
	Acquisition	8.7	0.3	6	6.90E-01	6.52E-01	0.64	0.911	1	4	30	1200	4.41E-01	
7 Corridor	Uptake (L)	15	0	0	1	1	0.64	0.832	1/5	6	60	600	1.37E-01	0.16
	Uptake (R)	15	3	10	1.14E-02	3.98E-01	0.64	0.832	1/5	1.5	60	600	9.94E-03	
	Acquisition	8.7	0.3	15	6.90E-01	1.95E-01	0.64	0.911	1/5	6	30	1200	1.17E-02	
10 Corridor	Uptake (L)	15	1	10	2.48E-01	3.98E-01	0.64	0.832	1/5	4	60	600	3.04E-02	0.07
	Uptake (R)	15	0	10	1	3.98E-01	0.64	0.83	1/5	7	60	600	4.00E-02	
	Acquisition	8.7	0.3	25	6.90E-01	4.19E-02	0.64	0.91	1/5	9	30	1200	1.12E-03	
13 Room control	Uptake (L)	15	0	0	1	1	0.64	0.83	1	5	60	600	9.86E-01	1.90
	Uptake (R)	15	1	15	2.48E-01	1.95E-01	0.64	0.83	1	4	60	600	7.45E-02	
	Acquisition	8.7	0.3	0	6.90E-01	1	0.64	0.91	1	3.6	30	1200	8.35E-01	

Table 3: Effective annual dose for public

Administered activity (A); thickness of lead (X_{Pb}); equivalent thickness of concrete (X_C); lead transmission factor (K_{Pb}); concrete transmission factor (K_C); patient transmission factor (K_{pat}); dose reduction factor (R_i); occupancy factor (T); distance source-interest point (D); operation time (t_{op}); annual number of patients (N); effective annual dose (E).

LOCATION	OPERACIÓN	A [mCi]	X_{Pb} [cm]	X_C [cm]	K_{Pb}	K_C	K_{pat}	R_i	T	D [m]	t_{op} [min]	N	E [μ Sv]	E tot [μ Sv]
4 Office	Uptake (L)	15	3	15	1.14E-02	1.95E-01	0.64	0.832	1	3	60	600	6.1	61.4
	Uptake (R)	15	0	20	1	9.04E-02	0.64	0.832	1	7	60	600	45.5	
	Acquisition	8.7	0	20	1	9.04E-02	0.64	0.911	1	12	30	1200	9.8	
8 Hall	Uptake (L)	15	0	20	1	9.04E-02	0.64	0.832	1	7	60	600	45.5	69.1
	Uptake (R)	15	3	15	1.14E-02	1.95E-01	0.64	0.832	1	3	60	600	6.1	
	Acquisition	8.7	0	20	1	9.04E-02	0.64	0.911	1	9	30	1200	17.5	
9 Courtyard	Uptake (L)	15	1	20	2.48E-01	9.04E-02	0.64	0.832	1/20	6	60	600	1.0	1.5
	Uptake (R)	15	1	20	2.48E-01	9.04E-02	0.64	0.832	1/20	10	60	600	0.3	
	Acquisition	8.7	0	20	1	9.04E-02	0.64	0.911	1/20	12	30	1200	0.6	
11 Quality control	Uptake (L)	15	1	30	2.48E-01	1.94E-02	0.64	0.832	1	6.5	60	600	2.8	9.4
	Uptake (R)	15	1	30	2.48E-01	1.94E-02	0.64	0.832	1	8	60	600	1.9	
	Acquisition	8.7	0	30	1	1.94E-02	0.64	0.911	1	8	30	1200	4.8	
12 Courtyard	Uptake (L)	15	1	20	2.48E-01	9.04E-02	0.64	0.832	1/20	2.5	60	600	5.5	5.3
	Uptake (R)	15	1	20	2.48E-01	9.04E-02	0.64	0.832	1/20	8	60	600	0.5	
	Acquisition	8.7	0	20	1	9.04E-02	0.64	0.911	1/20	12	30	1200	0.6	
15 Parking	Uptake (L)	15	0.3	20	6.90E-01	9.04E-02	0.64	0.832	1/20	14	60	600	0.5	6.4
	Uptake (R)	15	0.3	25	6.90E-01	4.19E-02	0.64	0.832	1/20	8	60	600	0.6	
	Acquisition	8.7	0.3	20	6.90E-01	9.04E-02	0.64	0.911	1/20	3	30	1200	6.8	
16 Cyclotron	Uptake (L)	15	0	50	1	9.00E-04	0.64	0.832	1	12	60	600	0.2	2.2
	Uptake (R)	15	0	50	1	9.00E-04	0.64	0.832	1	7	60	600	0.5	
	Acquisition	8.7	0	50	1	9.00E-04	0.64	0.911	1	3	30	1200	1.6	
17 Corridor	Uptake (L)	15	1	15	2.48E-01	2.00E-01	0.64	0.832	1/5	8	60	600	3.8	14.7
	Uptake (R)	15	1	15	2.48E-01	2.00E-01	0.64	0.832	1/5	11	60	600	2.0	
	Acquisition	8.7	0.3	15	6.90E-01	2.00E-01	0.64	0.911	1/5	7	30	1200	8.8	
18 Parking	Uptake (L)	15	0.3	20	6.90E-01	9.04E-02	0.64	0.832	1/20	13	60	600	0.5	51.0
	Uptake (R)	15	0.3	20	6.90E-01	9.04E-02	0.64	0.832	1/20	7	60	600	2.0	
	Acquisition	8.7	0	0	1	1	0.64	0.911	1/20	4	30	1200	61.2	

Since the effective annual doses calculated for all points are lower than the restrictions required by the standard AR10.1.1 both for workers and public, this installation complies with the requirements of shield design against external radiation set by the ARN.

4. Conclusions

The high energy of the annihilation photons of radionuclides used in PET technique together with long patient waiting times and image acquisition times make the requirements for setting up a PET facility different from those established for conventional nuclear medicine. The ARN methodology for the independent assessment of PET facilities consists of calculating the annual effective dose to workers and members of the public and comparing these with the design dose constraints established by this regulatory body.

The assessment carried out in *Fundación Centro Diagnóstico Nuclear* indicates that this facility complies with the requirements of the *Nuclear Regulatory Authority* in terms of shielding for external radiation.

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Radiation Protection in NORM Industries

Canoba, A.

Presentado en: 12th International Congress of the International Radiation Protection Association.
Buenos Aires, Argentina, 19-24 octubre 2008.

El presente trabajo se presentó oralmente durante el Seminario 2: Radiation Protection in NORM Industries.

RADIATION PROTECTION IN NORM INDUSTRIES: INTRODUCTION

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Abstract

NORM is an acronym for *naturally occurring radioactive materials*, which include radioactive elements found in the environment. Long-lived radioactive elements of interest include uranium, thorium and potassium, and any of their radioactive decay products, such as radium and radon. These elements have always been present in the earth's crust, and nearly all materials contain trace amounts of them. However, when these materials are processed as the result of human activities, concentration or enhancement of the levels of these radionuclides may occur. The processing of raw materials by many resource-based industries may increase the concentration of radioactive substances in those materials, and enhance the potential exposure to naturally occurring radioactive materials in products, by-products, residues and wastes. The most significant industries within the EU, based on the radiological risk and economic significance are: the phosphate industry, the processing of metal ores, zircon sands and refractory materials, manufacture of rare earths, manufacture and use of thorium compounds, the titanium dioxide pigment industry and the oil and gas extraction. If the residues containing naturally occurring radionuclides are not managed properly and safely, contamination over large areas is possible given the large quantities of such residues.

There are two effects of human exploitation that are relevant in the case of potential effects of NORM on human health and the environment:

- (1) The concentrations of NORM can be enhanced above its natural levels in a product, by-product or residue.
- (2) The availability for release into the biosphere of the NORM in products, by-products or residues can be enhanced through physicochemical changes or simply due to the method by which the residues are managed.

The pathways by which workers could receive a significant radiation dose are: external irradiation, inhalation of dust, inhalation of radon, ingestion of dirt and dust and skin contamination.

Occupational exposure to NORM falls within the scope of the requirements for practices if the radionuclide activity concentration in the material exceeds 1 Bq/g for uranium and thorium series radionuclides (other than radon and its short-lived progeny) or 10 Bq/g for potassium-40. If the relevant activity concentration level for radon or NORM is exceeded, a 'graded approach' to regulation should be applied, being the regulation in accordance with the characteristics of the operation and the exposures involved.

Radiological Protection of Patients in Nuclear Medicine

Rojo, A.M.

Presentado en: 12th International Congress of the International Radiation Protection Association.
Buenos Aires, Argentina, 19-24 octubre 2008.
El presente trabajo se presentó oralmente durante el Seminar 1: Radiological Protection of Patients.

RADIOLOGICAL PROTECTION OF PATIENTS IN NUCLEAR MEDICINE

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Argentina

This lecture aims at presenting the state of the art of radiological protection of patients in nuclear medicine focusing on three aspects of interest where to achieve improvement.

The hierarchy of the justification principle of the radiation protection is one of them. There seems for a change to be presented in the paradigm of the radiological protection of patients. The role of the physician who prescribes the medical practice becomes more relevant, together with the nuclear medicine specialist who should be co-responsible for the application of this justification principle. Regarding the doses optimization and the implementation of Dose Reference Level the involvement extends far beyond the physician and radioprotection officer. It is clear that the Medical Physicist is to play a very relevant role in the coordination of actions, as the nuclear medicine technician is to execute them.

Another aspect to consider is patient specific dosimetry. It should become a routine practice through calculation of the absorbed dose based on biodistribution data. It should be assessed for each individual patient, as it depends on a number of patient-specific parameters, such as gender, size and the amount of fatty tissue in the body, as well as the extent and nature of the disease.

In most cases, dosimetry calculations are not carried out and patients are administered standard levels of activity. There may be situations with a lack of knowledge on internal dosimetry as in many centers either none or only one or two medical physics experts are available. It shows that a formal training for experts in internal dosimetry at national level is required.

However up to now, there has been no satisfactory correlation between absorbed dose estimates and patient response. Moreover, the radiation protection for the patient is not assured, as the dose values given are often numbers without connection to radiobiological and/or hematological findings.

Pending tasks related to knowing the therapy successful rates, to comparing the dose values with radiobiological findings and to determine late radiation effects to the patient should be a goal for radiological protection of patients in nuclear medicine.

Finally, regarding regulations, it is interesting to discuss regulations, a balance should be achieved between: too many or not enough. The challenge for the regulator is to establish a regulatory structure that helps to minimize these unintended risks, while avoiding undue interference in medical judgments.

Too many regulations can result in medical or technological needless delay of while too few regulations can lead to an increase in misadministrations and increase the likelihood for patients, medical staff, and the public to receive involuntary exposures.

A careful balance must be attained in establishing an appropriate regulatory regime. Regulation should provide the appropriate vehicle for reducing radiation risks, while also minimizing interference with the beneficial uses of medicine.

Latin-American Biological Dosimetry Network (LBDNET) Intercomparison Exercise

Evaluation through triage and conventional scoring criteria.
Development of a new approach for statistical data analysis

Di Giorgio, M.; Vallergera, M.B.; Radl, A.; Taja, M.R.; Seoane, A.;
Stuck Oliveira, M.; Valdivia, P.; García Lima, O.; Lamadrid, A.;
González Mesa, J.; Romero Aguilera, I.; Mandina Cardoso, T.;
Guerrero Carvajal, C.; Arceo Maldonado, C.; Espinoza, M.;
Martínez López, W.; Di Tomasso, M.; Barquinero, F.; Roy, L.;
Lloyd, D.; Lindholm, C. and Romm, H.

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Buenos Aires, Argentina, 19-24 octubre 2008

El presente trabajo se presentó con poster y oralmente durante el evento.

LATIN-AMERICAN BIOLOGICAL DOSIMETRY NETWORK (LBDNET) INTERCOMPARISON EXERCISE

Evaluation through triage and conventional scoring criteria

Development of a new approach for statistical data analysis

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Abstract

Biological Dosimetry is a necessary support for National Radiation Protection Programs and Emergency Response Schemes. A Latin-American Biological Dosimetry Network (LBDNET) has been constituted by the biological dosimetry laboratories from: Argentina, Brazil, Chile, Cuba, Mexico, Peru, and Uruguay (IAEA Regional Project RLA9/054, 2007).

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The biological dosimetry laboratory of Argentina organized an international biological dosimetry intercomparison for the analysis of some relevant parameters involved in dose assessment, to reinforce the response capability in accidental situations requiring the activation of mutual assistance mechanisms and thus, constituting the bases of the LBDNET organization.

General Objectives

To evaluate the inter-laboratory reproducibility.

To identify potential difficulties and to promote the necessary modifications for the collaborative task.

Specific Objectives

To initiate, with this exercise, periodic intercomparisons among LBDNET participants.

To develop technical competence of associated laboratories inside the country.

INTERCOMPARISON PROGRAM

Participant laboratories from the LBDNET, Spain (UAB), France (IRSN), United Kingdom (HPA), Finland (STUK) and Germany (BfS).

ACTIVITIES: Are at present in progress and should be concluded in June 2008.

1. Human blood samples were irradiated *in vitro* with a ^{60}Co source (Regional Reference Center for dosimetry of the Atomic Energy Commission), two points of doses and a control. Cultures and slide preparations were performed according to standard methods.
2. Samples were distributed to the laboratories.
3. *Analyzing the samples:* In progress in each participant laboratory. The following data are required:
 - Number of dicentrics observed in 50 cells/30 dicentrics (triage scoring criteria), 100 cells (as an intermediate step) and 500 cells/100 dicentrics (conventional scoring criteria). In the case of two scorers: 250+250 cells each; reporting raw data, frequency and standard deviation associated to the observations.
 - Dose estimates with 95% confidence limits, reporting the calibration curve applied (coefficients and its standard deviations).
 - Distribution of aberrations in the analyzed cells, relative variance, Papworth *u* test index.
4. *Statistical analysis:* The presence of individual laboratories or values that appear to be inconsistent with all other laboratories may change the estimates and thus, decisions have to be taken. To discard or correct inconsistent values, two approaches will be used (ISO 5725-2 /5):
 - Numerical outlier tests (Cochran and Grubbs tests): To discard data that give rise to a test statistic that exceeds the critical value of the test at the 1% significance level.
 - Robust methods for data analysis: To yield robust values of the average and standard deviation of the data.

5. *Estimating the mean value and the standard deviation inter-laboratory:* To estimate the parameters once outliers are discarded or corrected.
6. *Determining laboratory's performance:* To calculate z-score parameter from the laboratory results, the reference value and the estimated standard deviation. To determine u-score parameter, which evaluation includes both participant measurements and reference value uncertainties.
7. *Reporting results:* To allow to implement corrective actions to reinforce service capabilities of the laboratories.

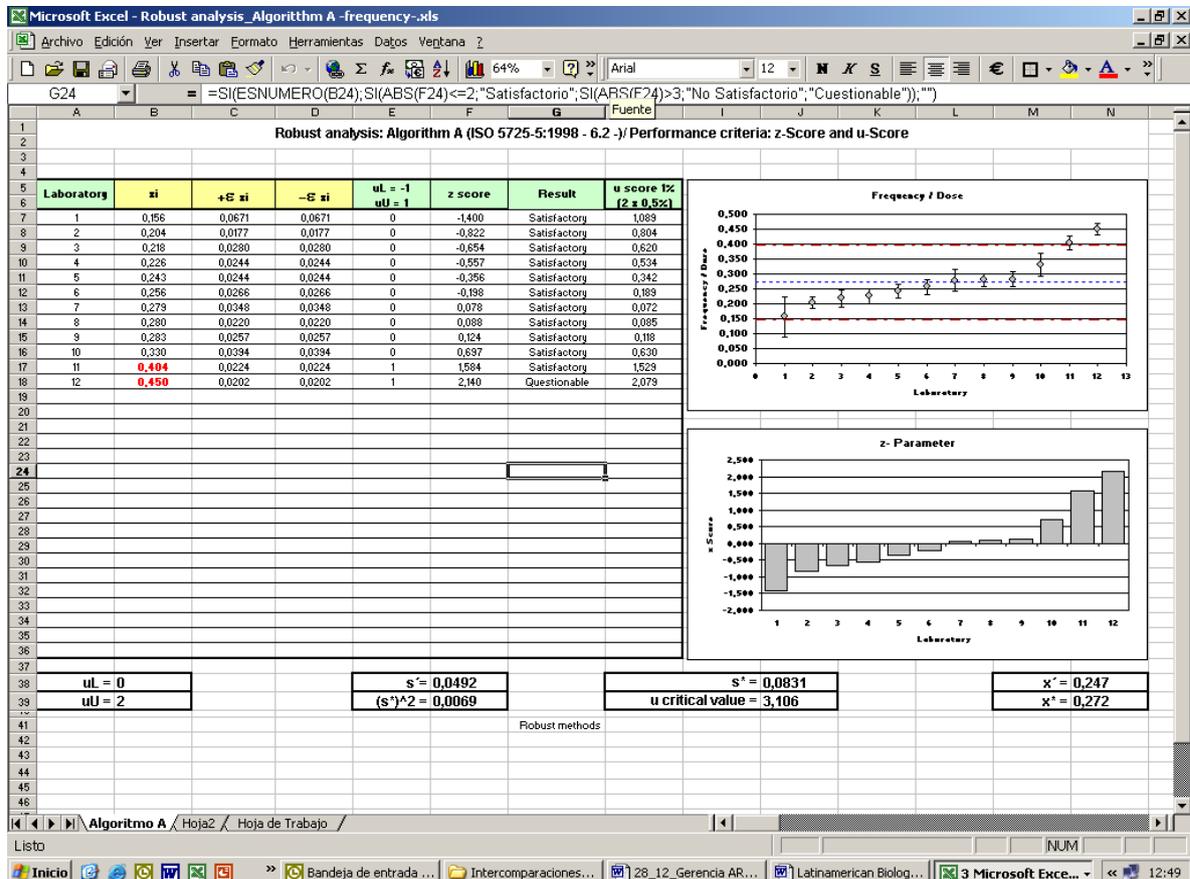
PRELIMINARY CONCLUSIONS: Proficiency tests are essential tools for the quality assurance of the laboratories as they constitute an objective evaluation of its performance, from both human and technical point of view. Actions for a steady quality assurance and quality control, such as this Intercomparison exercise, will be the technical support of the LBDNET. Additionally, it will contribute in the accrediting process (ISO 19238:2004 and ISO/IEC 17025:2005 requirements).

Keywords: Biological Dosimetry; Intercomparison Exercise; Triage; Statistical Data Analysis

The participation of Latin- American biological dosimetry laboratories in: Goiania (Brazil) and El Salvador accidents, and more recently, Cochabamba (Bolivia) and Concepcion (Chile) accidents, and Venezuela incidents show that the laboratories have reached a common background level in the organization and in the response in cases of overexposures to IR that have allowed to formalize a LBDNET for cooperation purposes. Two important tasks of the present intercomparison exercise are:

1. Evaluation through triage and conventional scoring criteria: The required data focus on cytogenetic triage (to evaluate approximately and rapidly the radiation doses received by individuals in order to support the clinical categorization of casualties) and conventional dosimetry. The assessment (correlation) of these results would allow to establish the applicability of triage criteria in emergency situations.

2. Development of a new approach for statistical data analysis: A software and Excel tools for data analysis were developed and validated by the organizer and were agreed to by the participant parties (during a meeting held in La Havana - Cuba, November 2007 in the frame of the IAEA Regional Project RLA9/054). It was also decided: a) to use robust methods (Algorithm A). This algorithm yields robust values of the average and standard deviation of the data to which is applied. When robust methods are used, the outlier tests and consistency checks described in ISO 5725-2 or-5 should be applied to the data, and the causes of any outlier should be investigated; 2) to apply z- scores and u- scores for determining the laboratory's performance; u-scores includes uncertainties of the participant measurements and the uncertainty of the reference value. Laboratories performing well in classical proficiency testing will not necessarily exhibit the same level of performance when their analytical uncertainties are considered in the evaluation.





Latin American Biological Dosimetry Network Intercomparison Exercise

Evaluation through triage and conventional scoring criteria

Development of a new approach for statistical data analysis



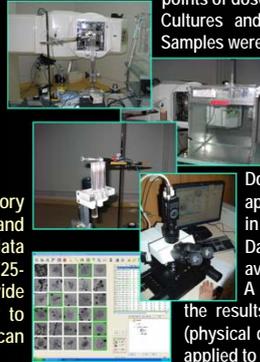
M. Di Giorgio, M. Vallergera, A. Radl, M. Taja, A. Seoane, M. Stuck Oliveira, P. Valdivia, O. García Lima, A. Lamadrid, J. González Mesa, I. Romero Aguilera, T. Mandana Cardoso, C. Guerrero Carvajal, C. Arceo Maldonado, M. Espinoza, W. Martínez López, M. Di Tomasso, H. Romm, F. Barquiner, C. Lindholm, L. Roy, I. Güçlü, D. Lloyd - Argentina-ARN, Argentina-UNLP, Brazil-IRD, Chile-CCHEN, Cuba-CPHR, Mexico-ININ, Peru-IPEN, Uruguay-IIBCE, Finland-STUK, France-IRSN, Germany-BIS, Spain-UAB, Turkey-CNRTC, United Kingdom-HPA

INTRODUCTION

Biological Dosimetry is a necessary support for National Radiation Protection Programs and Emergency Response Schemes. A Latin-American Biological Dosimetry Network (LBDNET) has been constituted by the laboratories from: Argentina, Brazil, Chile, Cuba, Mexico, Peru and Uruguay (IAEA - RLA9/054, 2007). The biological dosimetry laboratory of Argentina organized an international biological dosimetry intercomparison, involving 7 Latin American and 6 European countries, for the analysis of some relevant parameters related with dose assessment, to reinforce the response capability in accidental situations requiring the activation of mutual assistance mechanisms and thus, constituting the bases of the LBDNET organization.

Objectives:

To evaluate the inter-laboratory reproducibility and intra-laboratory repeatability. To assess the results derived from conventional and triage scoring criteria. To apply an statistical approach for data analysis and performance evaluation based on ISO 43-1:1997, 5725-5:1998 and 13528:2002 standards. To identify difficulties and provide the necessary modifications for the collaborative task in order to improve the technical quality and competence of Latin American laboratories.



MATERIALS AND METHODS

INTERCOMPARISON PROGRAM

Participant laboratories from the LBDNET, Finland, France, Germany, Spain, Turkey and United Kingdom. Human blood samples were irradiated in vitro with a ⁶⁰Co source (Regional Reference Center for dosimetry of the Atomic Energy Commission), two points of doses and a control.

Cultures and slide preparations were performed according to standard methods. Samples were distributed to the laboratories.

Data required: Number of dicentric chromosomes observed in 50 cells/30 dicentrics (triage scoring criteria), 100 cells and 500 cells/100 dicentrics (conventional scoring criteria). For intra-laboratory evaluation (2 scorers): 250+250 cells each; reporting raw data, frequency and standard deviation associated to the observations.

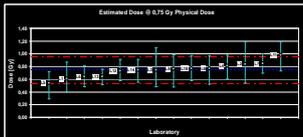
Dose estimates with 95% confidence limits, reporting the calibration curve applied (coefficients and its standard deviations). Distribution of aberrations in the analyzed cells, relative variance, Papworth u test index.

Data analysis: Robust methods were applied to yield robust values of the average and standard deviation of the data.

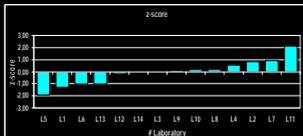
A consensus value for inter-laboratory frequency results (robust average of the results reported by the participants) and a reference value for dose results (physical dose) were determined. Performance statistics (Z-score and U-score) were applied to the data.

RESULTS AND DISCUSSION

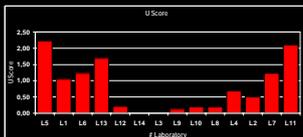
DI: 0.75 Gy – Reproducibility (Algorithm A) 500 cells



Dose values determined by the laboratories with their expanded uncertainties
 $x^* \pm 1.96s^*$ and $x^* - 1.96s^*$
 $x^* \pm s^* [Gy] = 0.744 \pm 0.109$
 Physical dose [Gy] = 0.75 ± 0.023
Precision
Trueness
Accuracy purpose is met

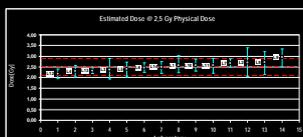


$Z = (x_i - x_{ref}) / S^*$
 $|Z| \leq 2$ satisfactory
 $2 < |Z| < 3$ questionable - "warning signal"
 $|Z| \geq 3$ unsatisfactory - "action signal"
L11: questionable

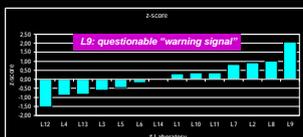


$U = |x_{ref} - x_i| / (u_{ref}^2 + u_i^2)^{1/2}$
U Score: the uncertainty of the participant's measurements result is taken into account for the evaluation of performance
 U critical value: 3.0123
 If $u > 3.0123$, the values differ significantly

DII: 2.50 Gy - Reproducibility (Algorithm A) 500 cells



Dose values determined by the laboratories with their expanded uncertainties
 $x^* \pm 1.96s^*$ and $x^* - 1.96s^*$
 $x^* \pm s^* [Gy] = 2.515 \pm 0.204$
 Physical dose [Gy] = 2.5 ± 0.075
Precision
Trueness
Accuracy purpose is met



$Z = (x_i - x_{ref}) / S^*$
 $|Z| \leq 2$ satisfactory
 $2 < |Z| < 3$ questionable - "warning signal"
 $|Z| \geq 3$ unsatisfactory - "action signal"
L9: questionable "warning signal"

Repeatability (Algorithm S) 500 cells

This algorithm is applied to intra-laboratory standard deviations. It yields a robust pooled value of the standard deviations to which is applied.

w^* = robust standard deviation of $n=2$ replicate measurements.

S_r = robust repeatability standard deviation = $w^* / (n)^{1/2}$ (ISO 5725-5 clause 6.4.1)

S_R = reproducibility standard deviation (ISO 5725-5 clause 6.4.2.3)

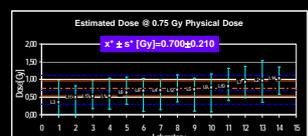
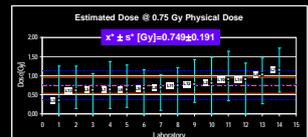
8 laboratories presented data for this study

Parameter	DI		DII	
	Frequency	Dose	Frequency	Dose
S_r	0.0065	0.0590	0.0364	0.1214
S_R	0.0127	0.1164	0.0502	0.2210

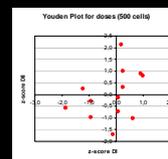
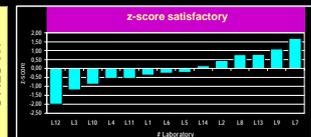
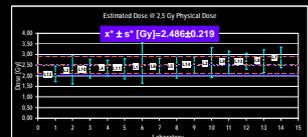
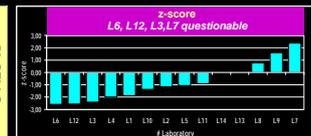
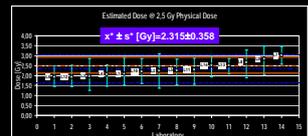
✓ The reproducibility (inter-laboratory) standard deviation resulted 22% for DI and 11.6% for DII

✓ S_R and S_r for frequencies presented lower values than those for doses

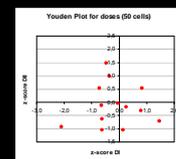
DI: 0.75 Gy - Cytogenetic triage



DII: 2.5 Gy - Cytogenetic triage



Youden Plot: Laboratory errors seem to be random



CONCLUSIONS

➤ Robust methods (Algorithms A and S) were applied to calculate repeatability and reproducibility standard deviations:

➤ S_R resulted 22% for DII and 11.6 % for DI. S_r resulted greater for dose respect to frequency, indicating a higher dispersion in dose assessment associated with the different criteria applied to convert frequency into dose and to the diversity of calibration curves applied.

➤ Intra-laboratory indicated lower dispersion. S_r resulted lower for frequencies than for doses.

➤ An increased dispersion with dose was observed as expected due to the complexity.

➤ Triage criteria: The evaluation of 50 cells (DI) revealed the probability to obtain false negatives as the confidence interval (CI) reaches zero dose in 6 cases. When analyzing 100 cells the CI tended to detach from zero.

➤ Overall, the laboratory performance was satisfactory for mutual cooperation purposes.

➤ Round exercises are suggested to maintain and reinforce the response capacity of the laboratories.

Normal Tissue Adverse Side Effects in Radiotherapy Cancer Patients and Applicability of Predictive Radiosensitivity Tests for New Radiation Treatment Decision

Di Giorgio, M.; Sardi, M.; Vallerga, M.B. and Radl, A.

Presentado en: 12th International Congress of the International Radiation Protection Association.
Buenos Aires, Argentina, 19-24 octubre 2008
El presente trabajo presentó poster durante el evento.

NORMAL TISSUE ADVERSE SIDE EFFECTS IN RADIOTHERAPY CANCER PATIENTS AND APPLICABILITY OF PREDICTIVE RADIOSENSITIVITY TESTS FOR NEW RADIATION TREATMENT DECISION

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Abstract

Around 5%-7% of cancer patients develop adverse side effects, which include acute effects, late effects and cancer induction to radiation therapy in normal tissues in the treatment field. Such effects are of particular interest as the cancer patient population that reaches prolonged survival has increased with the improvements in cancer therapy and health care. These adverse reactions are mainly influenced by deficiencies in DNA repair pathways. However, tissue response to IR could be modified by several treatment- and patient- related factors.

Numerous studies have been carried out to evaluate the correlation between clinical and cellular radiosensitivity, by in vitro tests. Previous own studies, characterizing DNA repair capacity in peripheral lymphocytes of cancer patients through cytokinesis blocked micronucleus test and alkaline single-cell microgel electrophoresis (comet), indicated that such assays correlated with the clinical radiation signs of radiosensitivity and showed the predictive potential of both techniques in the identification of radiosensitivity subgroups.

In this paper, retrospective studies are conducted in 10 representative cases, which had developed acute or late toxicity in previous treatments and at present require new radiation treatments due to secondary malignancies or recurrence.

Samples 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 (0-120 minutes). Captured images were analyzed by CASP image analysis software. Repair capacity was quantified by the Olive tail moment. Weibull alpha parameter was applied to describe DNA damage at the different evaluated repair times after in vitro irradiation and fitted by a mono-exponential model to describe the kinetic profile.

In every evaluated patient a correlation between mean half-time ($T_{1/2}$) and residual damage (**RD**) parameters was observed. Whereas, one case, revealed a lack of correlation between both parameters: average reactor according to $T_{1/2}$ and over reactor according to **RD**. This observation made us aware of the importance of **RD** parameter in the evaluation of individual radiosensitivity. In this case, the patient presented a 46% of non-repair initial DNA damage and so qualified as a patient with a greater risk than average of developing radiation toxicity.

Thus, the fundamental understanding of $T_{1/2}$ (the speed of repair in the mono-exponential repair kinetic model applied) and the **RD** (percentage of the lymphocytes that do not restore the induced initial DNA damage) constitute important parameters for the individual radiosensitivity analysis. Moreover, when analyzing Weibull distributions

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at each repair time, patients that showed two cellular subpopulations (damaged and undamaged cells) resulted in normal/average radiosensitivity of the lymphocyte pool and therefore, considered as patients with normal/average recovery from the cytogenetic induced damage. MN and comet techniques enable to characterize the cytogenetic status of patients that had developed acute or late toxicity in previous treatments. The differences between average and over reactors were significant. These assays showed a good predictive potential for the detection of patients with a greater risk than average patients of developing radiation toxicity.

KEYWORDS: Predictive Radiosensitivity Tests; DNA Repair Capacity; Radiotherapy; Adverse Side Effects

Patients	Clinical and <i>in vitro</i> radiosensitivity – $T_{1/2}$			
	Mean half-time $T_{1/2}$ [min]			Residual Damage TM_R
	Healthy controls (2,6±0,3)	Average-reactor patients (4,7±2,9)	Over-reactor patients (24,9±10,4)	
A. B.			9,42 ± 2,97	0,13±0,47
M. A. G.			14,52 ± 8,40	14,9±6,12 0,24±0,10 (normalized value)
P. M.		6,0±4,2		27,35±2,73 0,64±0,06 (normalized value)
H. A. C.			22,8±5,7	0,58±0,45 0,09±0,07 (normalized value)
D. G.		6,7±2,0		5,5±1,7 0,19±0,06 (normalized value)
R. M.	1,9±0,6			3,2±1,6 0,09±0,04 (normalized value)
O. T.	0,9± 0,2			3,5±0,2 0,16±0,01 (normalized value)
I.	1,3± 0,2			0,6±0,2 0,03±0,01 (normalized value)
C.	In progress			
P.A.	In progress			

Table 1. Comet assay results of the patients evaluated retrospectively

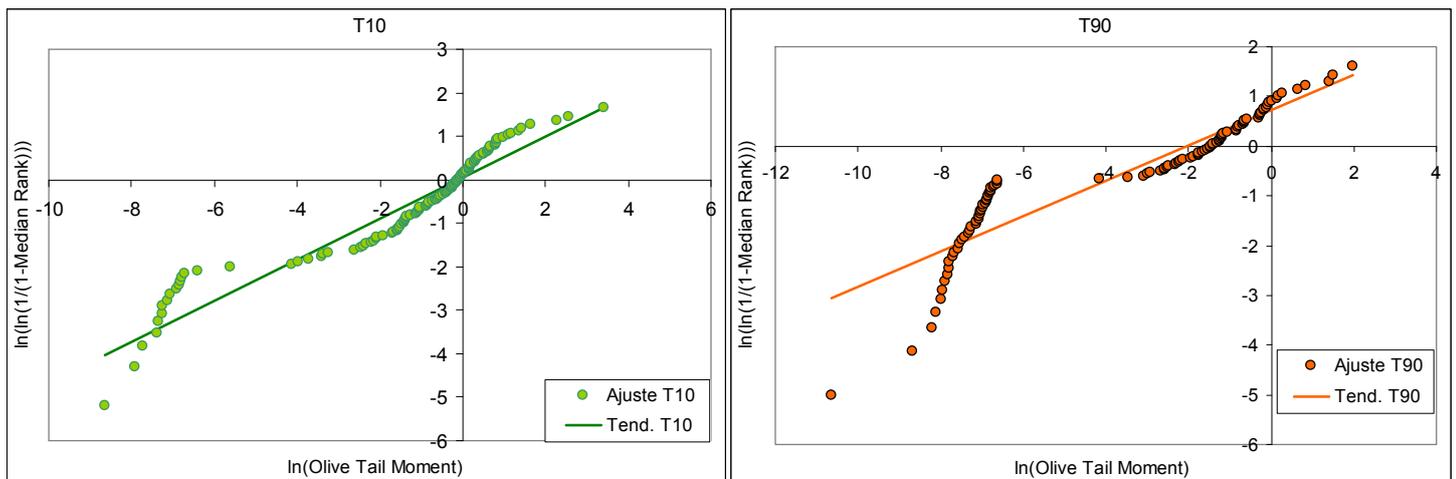


Figure 1. Weibull distributions at different repair times showing two cellular subpopulations

NORMAL TISSUE ADVERSE SIDE EFFECTS IN RADIOTHERAPY CANCER PATIENTS AND APPLICABILITY OF PREDICTIVE RADIOSENSITIVITY TESTS FOR NEW RADIATION TREATMENT DECISION



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¹Autoridad Reguladora Nuclear-²Hospital Italiano-³Mevaterapia
 Buenos Aires – Argentina

INTRODUCTION

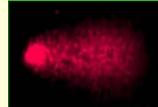
Adverse side effects (acute, late and cancer induction) to radiotherapy in normal tissues, unavoidably included within the treatment field, are of particular interest as the cancer patient population that reaches prolonged survival has increased with the improvements in cancer therapy and health care. Deficiencies in DNA repair pathways influence its development. However, tissue response to IR could be modified by several treatment- and patient-related factors.

Numerous studies have been carried out to evaluate the correlation between clinical and cellular radiosensitivity, by *in vitro* tests. Previous own studies, characterizing DNA repair capacity in peripheral lymphocytes of cancer patients through cytokinesis blocked micronucleus test and alkaline single-cell microgel electrophoresis (comet), indicated that such assays correlated with the clinical radiation signs of radiosensitivity and showed the predictive potential of both techniques in the identification of radiosensitivity subgroups.

OBJECTIVE: To evaluate retrospectively DNA repair capacity and its correlation with clinical response in 8 representative cases, which had developed acute or late toxicity in previous treatments and at present require new radiation treatments due to recurrence or second tumor, patients with comorbidities and patients with non expected responses to the administrated radiotherapy, for treatment decisions.

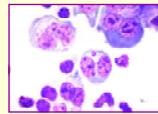
MATERIALS AND METHODS

- Blood samples 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 (0-120 minutes). Captured images were analyzed by CASP image analysis software. Repair capacity was quantified by the Olive tail moment. Weibull alpha parameter was applied to describe DNA damage at the different evaluated repair times after *in vitro* irradiation and fitted by a mono-exponential model to describe the kinetic profile.



$$DTM(t) = DTM_0 \cdot e^{-\left[\frac{t \cdot \ln 2}{T_{1/2}}\right]} + DTM_R$$

DTM = Damage (Olive tail moment) $T_{1/2}$ = Repair half time
 DTM_0 = Initial damage t = Incubation time after *in vitro* irradiation
 DTM_R = Residual damage



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-reactor cancer patients (24.9 ± 10.4 minutes). Differences among means resulted significant, verified with Student's t-test.

RESULTS AND DISCUSSION

DNA repair capacity – Clinical response

Patients	Clinical and <i>in vitro</i> radiosensitivity				Rationale
	Mean half-time T1/2 (min)			Residual Damage TM_R	
	Healthy controls	Average – Reactors patients	Over-reactor patients		
Case 1 recurrence breast Ca		9.42 ± 2.97		0.13 ± 0.47	Infectious vs.actinic mastitis
Case 2 2nd tumor			14.52 ± 8.40	0.24 ± 0.10	Actinic pneumonitis
Case 3 lung tumor - CNS metastasis		6.0 ± 4.2		0.64 ± 0.06	Actinic pneumonitis Acute toxicity to CNS
Case 4 Buttock liposarcoma - Mediastinal metastasis			22.8 ± 5.7	0.09 ± 0.07	Acute epithelitis
Case 5 Rectal NHL		6.7 ± 2.0		0.19 ± 0.06	Rectitis Haemorrhoidal syndrome
Case 6 Rectal adenocarcinoma	1.9 ± 0.6			0.09 ± 0.04	Diabetes – hypothyroidism -collagenopathy
Case 7 Larynx Ca	0.9 ± 0.2			0.16 ± 0.01	Late toxicity Hypo-fractionation
Case 8 Prostate Ca	1.3 ± 0.2			0.03 ± 0.01	Rectal and urinary toxicity; asthenia

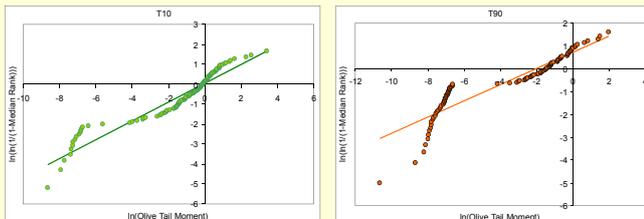
Case 1 follow-up

Patient developing a supposed actinic mastitis that led to treatment break during her first radiotherapy. During her second treatment poor acute toxicity (grade 1) and non late toxicity were observed, correlating with radiosensitivity test results



T1/2 and RD parameter correlation

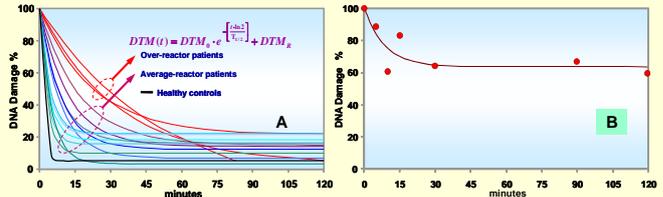
- ✗ Taken together, these results suggest a correlation between mean half-time (T1/2) and residual damage (RD) parameters.
- ✗ Whereas, one patient (case 3), revealed a lack of correlation between both parameters: average reactor according to T1/2 and over reactor according to RD. This observation made us aware of the importance of RD parameter in the evaluation of individual radiosensitivity. In this case, the patient presented a 64% of non-repair initial DNA damage and so qualified as a patient with a greater risk than average of developing radiation toxicity.



Weibull distributions at different repair times showing two cellular subpopulations

What happens at the cellular population level?

- Assuming two types of patient responses (normal/average radiosensitivity and hypersensitivity), it was observed an association between homogeneity/heterogeneity of the analyzed cellular population and radiosensitivity patient response:
- ✓ Patients with two cellular subpopulations (damaged and undamaged cells) resulted in normal/average radiosensitivity of the lymphocyte pool, whereas patients with one predominant cellular subpopulation (damaged cells) resulted in hyper-radiosensitivity.



- ✗ DNA repair kinetic profiles showing: A) correlation between T1/2 (the speed of repair in the mono-exponential kinetic model) and RD (percentage of the lymphocytes that do not restore the induced initial DNA damage) parameters and B) non-correlation in case 3

CONCLUSIONS

- MN and comet techniques enable to characterize the cytogenetic status of patients that had developed acute or late toxicity in previous treatments, patients with comorbidities and patients with unexpected severe toxicities to the administrated radiation treatment.
- These assays showed a good predictive potential for the detection of patients with a greater risk than average patients of developing radiation toxicity.
- The fundamental understanding of T1/2 and the RD constitute important parameters for the individual radiosensitivity analysis.
- A correlation between cellular subpopulation composition (damage and undamaged cells) and patient radiosensitivity status was observed.

Reproductive Function and Biological Dosimetry Prospective Study of Young Thyroid Differentiated Cancer Patients Treated with I-131

Di Giorgio, M.; Chebel, G.; Fadel, A.M.;
Vallerga, M.B.; Kundt, M.; Taja, M.R.; Radl, A;
Gutiérrez, S.; Normandi, E. and Levalle, O.

Presentado en: 12th International Congress of the International Radiation Protection Association.
Buenos Aires, Argentina, 19-24 octubre 2008
El presente trabajo se presentó con poster y oralmente durante el evento.

**REPRODUCTIVE FUNCTION AND BIOLOGICAL DOSIMETRY
PROSPECTIVE STUDY OF YOUNG THYROID DIFFERENTIATED CANCER
PATIENTS TREATED WITH I-131**

¹Marina Di Giorgio* ; ²Graciela Chebel; ²Ana María Fadel; ¹María Belén Vallerga; ³Miriam Kundt; ¹María Rosa Taja; ¹Analfía Radl; ²Silvia Gutiérrez; ²Eduardo Normandi; ²Oscar Levalle

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Abstract

The administration of I-131 in the management of differentiated thyroid cancer (DTC) is a well established practice. As the spermatogonia is highly sensitive to radiation, large doses of internal radiation could result in adverse effects on reproductive function such as oligo/azoospermia and infertility.

During spermiogenesis, mammalian chromatin undergoes replacement of nuclear histones by protamines, which yields a DNA sixfold more highly condensed in spermatozoa than in mitotic chromosomes. The structure of this highly packaged chromatin shows a low binding capacity for several fluorochromes and dyes such as chromomycin A₃ (CMA₃).

The aim of this study is to assess the correlation between reproductive function (endocrine and exocrine testicular function, and levels of CMA₃ stainability) and biological dosimetry in a prospective study of 4 young DTC patients treated with I-131.

In this context, a background level of CMA₃ binding in mature human sperm was established. It revealed a variable accessibility of CMA₃ to the DNA that is dependant on packaging quality and thus, indicative of protamine deficiency. The identification of altered stainability suggests DNA damage as well as epigenetic effects, which may be indicators of male infertility.

Transient impairment of spermatogenesis associated with an increase in FSH, an altered spermiogram and even azoospermia was observed after the administration of cumulative activities. Overall, testosterone levels were preserved, except in one case, which presented a drastically diminished value associated with an increase in LH level.

As peripheral blood lymphocytes and spermatogonia have equivalent radiosensitivity (interphase death) we hypothesize that the knowledge of DNA damage recovery in peripheral lymphocytes could correlate with spermatogonia recovery and with FSH evolution.

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Therefore, a prospective study on the decline of unstable chromosome aberrations is being conducted, considering the damage induced by each therapeutic administrated activity and its respective decline before a new required administration.

KEYWORDS: Differentiated thyroid cancer; I-131 therapy; reproductive function; chromomycin A3; biological dosimetry

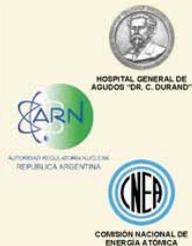
Patient	Age [years]	Total Activity [mCi]	Time from last dose [months]	LH [mIU/ml]	FSH [mIU/ml]	Testosterone [ng/ml]	Testo/LH [nmol/IU]	Sperm concentration [Mill/ml]
I	26	150	5	6.4	8.6	6.5	3.5	1.8
		450						Azoospermia
II	28	650	16	28.0	7.4	1.7	0.2	Azoospermia
III	33	450	9	4.5	12.6	4.5	3.4	< 1.0
		750	12	4.6	16.4	4.8	3.6	Azoospermia
			20	3.7	12.5	5.1	4.8	-
			24	-	10.3	-	-	15.0
IV	35	200	9	8.4	23.3	3.4	1.4	Azoospermia
			11	5.2	21.7	4.0	2.6	1.0
			14	11.0	19.3	-	-	2.4
			17	-		-	-	58.0
			27	6.3	10.2	-	-	45.0
			45	-		-	-	Pregnancy

Table 1: Data from the evaluated patients. Semen parameters and blood levels of sexual hormones.

Sperm concentration, normal range 20 to 250 million/ml (WHO); FSH: normal range 1.5 to 7.0 mIU/ml;

LH: normal range 1.1-12.0 mIU/ml; Testosterone, normal range 3.0-9.0 ng/ml.

REPRODUCTIVE FUNCTION AND BIOLOGICAL DOSIMETRY PROSPECTIVE STUDY OF YOUNG THYROID DIFFERENTIATED CANCER PATIENTS TREATED WITH ¹³¹I



¹M. Di Giorgio*; ²G. Chebel; ²A.M. Fadel; ¹M.B. Vallerga;

³M. Kundt; ¹M.R. Taja; ¹A. Radl; ²S. Gutiérrez; ²E. Mormandi; ²A. Oneto; ²O. Levalle

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INTRODUCTION

The administration of ¹³¹I in the management of differentiated thyroid cancer (DTC) is a well established practice. As the spermatogonia is highly sensitive to radiation, large doses of internal radiation could result in adverse effects on reproductive function such as oligo/azoospermia and infertility.

0.15 Gy acute exposure → transient infertility
> 6 Gy acute exposure → permanent infertility

During spermiogenesis, mammalian chromatin undergoes replacement of nuclear histones by protamines, which yields a DNA highly condensed in spermatozoa. The structure of this highly packaged chromatin shows a low binding capacity for several fluorochromes such as chromomycin A3 (CMA3).

Biological Dosimetry prospective studies conducted on samples from patients with cumulative doses, before and after each therapeutic administration, allows to evaluate DNA damage and repair capacity in peripheral blood lymphocytes.

OBJECTIVE: To assess the correlation between: reproductive function (endocrine and exocrine testicular function), levels of CMA3 stainability and biological dosimetry in a prospective study of 5 young DTC patients treated with ¹³¹I.

Dose to testes (euthyroid):
26-37 μGy/MBq of ¹³¹I -
ICRP 30 - Vol 23:3-1999
360 mCi = 0.33 Gy

MATERIALS AND METHODS

Patients: 5 young (26-35 years) DTC patients from C. Durand Hospital treated with total thyroidectomy, followed by ¹³¹I and levothyroxine therapy, with cumulative activities of 3.7 GBq to 30.9 GBq (100-835 mCi) for the treatment of thyroid remnants or metastases were assessed, with a follow-up up to 45 month after their last administrated activity.

Hormone measurements: Blood levels of LH, FSH and testosterone were assessed. FSH: NV 1.5 to 7.0 mIU/ml; LH: NV 1.1-9.0 mIU/ml; Testosterone, NV 3.0-9.0 ng/ml.

Spermiogram: Sperm concentration, NV 20-250 million/ml (WHO), percentage of motile sperm and percentage of sperm with normal morphology were evaluated.

CMA3: Non-irradiated semen samples from 35 men attending a Clinic of Sterility, 7 healthy controls and 2 DTC patients were evaluated by CMA3 staining. At least 500 spermatozoa were randomly evaluated for each sample. Staining pattern: bright green fluorescence (chromatin packaging abnormal) and dull green staining (normal).

Biological Dosimetry: a prospective study on the decline of unstable chromosome aberrations was conducted, considering the damage induced by each therapeutic administrated activity and its respective recovery before a new required administration. A blood sample was obtained before each patient treatment and another was obtained on day 8 after the ¹³¹I administration. Cytogenetic methods were applied to quantify chromosome aberrations, which were referred to a dose-response curve.

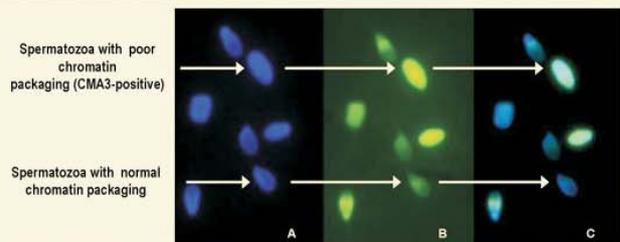
RESULTS AND DISCUSSION

Therapeutic administration, blood levels of sexual hormones and semen parameters

Patient	¹³¹ I Total Activity [mCi]	Last Administrated Activity (AA) [mCi]	Time post AA [month]	LH mIU/ml NV:1.1-9.0	FSH mIU/ml NV:1.5-7.0	Testo ng/ml NV:3-9	Sperm concentration [mill/ml] NV:20-250
I	150	150	5	6.4	8.6	6.5	1.8
	450	300	6	9	12.4	8.2	Azoospermia
	750	300	6	8.3	17	5.9	Azoospermia
II	850	250	16	10.1	18.8	4.8	Azoospermia
	835	185	12	4.7	22.4	3	Azoospermia
III	450	150	9	4.5	12.6	4.5	< 1
	750	300	12	4.6	16.4	4.8	Azoospermia
			20	3.7	12.5	5.1	-
			24	-	10.3	-	15
IV	200	100	9	8.4	23.3	3.4	Azoospermia
			11	5.2	21.7	4	1
			14	11	19.3	-	2.4
			17	-	-	-	58
			27	6.3	10.2	-	45
V	100	100	7	3.7	2.9	5.6	44
			12	6.5	12.7	4.9	7
			-	-	-	-	90 Asthenoteratozoospermia

- Transient impairment of spermatogenesis associated with an increase in FSH, an altered spermiogram and even azoospermia was observed after the administration of cumulative activities
- Overall, testosterone levels were preserved
- A drastic decrease in the spermatoc count (oligo/azoospermia) was observed 5 to 16 months after ¹³¹I treatment.
- Cases III and IV recovered spermiogenesis after 24 and 17 month after their last administrated activity respectively, reaching the latter pregnancy at term
- Gonadal function: the hypothalamo-hypophysial axis responded to the peripheral failure

Percentage of human spermatozoa positive (after treatment with CMA3)



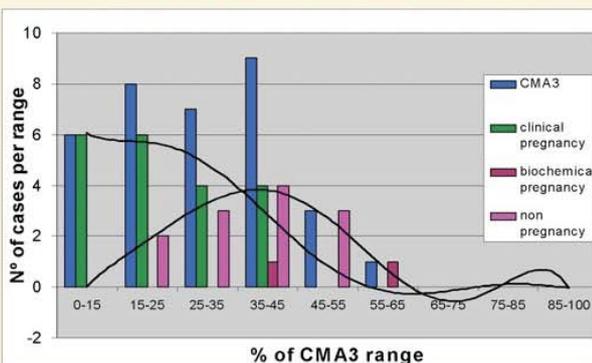
Sperm samples from ¹³¹I-treated patients showing fixed and DAPI stained spermatozoa for nuclear morphology observation (A), CMA3 staining for chromatin packaging analysis (B) and combination of both images (C)

- Hypothesis: The presence of DNA damage in mature spermatozoa correlates to poor chromatin packaging (≥ 30% of spermatozoa are CMA3-positive)
- As CMA3 binds to histones, spermatozoa positive indicates protamine deficit

PATIENT	SERUM CONCENTRATION [mill/ml]	CMA3 (+) NV: < 30%	TOTAL ACTIVITY [mCi] / TIME FROM LAST ADMINISTRATION [month]	BIOLOGICAL DOSIMETRY Recovery from last administration
I	AZOOSPERMIA	-	750/6	Non significant
II	AZOOSPERMIA	-	850/12	Non significant
III	84	13.6	750/35	Significant
V	90	19.5	100/12	Significant

- As peripheral blood lymphocytes and spermatogonia have equivalent radiosensitivity (interphase death) we hypothesize that the knowledge of DNA damage recovery in peripheral lymphocytes could correlate with spermatogonia recovery and with FSH evolution

Relationship between the percentage of human spermatozoa positive (after treatment with CMA3) and fertility



- Spermatozoa showed values of sensitivity to CMA3 ranging from 11% to 65% of total cells
- In vivo fecundity decreases progressively when ≥ 30% of the spermatozoa are identified as having DNA damage (positive CMA3, defective chromatin packaging)

CONCLUSIONS

- Alteration of reproductive function: transient impairment of spermatogenesis associated with an increase in FSH, preserved testosterone levels and an altered spermiogram and even azoospermia, was observed after ¹³¹I therapy for DTC, particularly after high levels of delivered activity.
- The identification of altered CMA3 stainability suggests DNA damage as well as epigenetic effects, which may be indicators of male infertility.
- Time required for lymphocytes to restore the damage induce by each therapeutic administration is dependant on the cumulative dose and correlates with CMA3 and sperm concentration.
- For patients particularly at risk for ¹³¹I-induced testicular damage, a long-term preservation of semen, obtained before ¹³¹I therapy, should be considered.
- Efforts to reduce some of the radiation sources to testes should be made by: keeping patients well hydrated with frequent urination and ensuring 1-2 bowel movements per day during the first 2-4 days after treatment.

Study Protocol for an Approach Based on Diagnosis and Therapy of Cutaneous Radiation Induced Lesions

Di Giorgio, M.; Portas, M.; Vallerga, M.B. and Radl, A.

Presentado en: 12th International Congress of the International Radiation Protection Association.
Buenos Aires, Argentina, 19-24 octubre 2008
El presente trabajo se presentó con poster y oralmente durante el evento.

STUDY PROTOCOL FOR AN APPROACH BASED ON DIAGNOSIS AND THERAPY OF CUTANEOUS RADIATION INDUCED LESIONS

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Abstract

In the frame of an agreement between the “Hospital de Quemados del Gobierno de la Ciudad de Buenos Aires”-Burn Center- (a reference hospital of the Medical Radiological Emergency Response Network of Argentina) and the Nuclear Regulatory Authority, a research project for diagnostic and therapeutic approach of cutaneous radiation syndrome (CRS) is in progress.

Sixty seven persons, which developed acute and/or late CRS, were included in this protocol from 1997 to 2007, treated with an equivalent therapeutic scheme and evaluated through clinical follow-up, serial photographic record and complementary tests (telethermography and high frequency ultrasonography).

There exist individual variations that could condition the response to ionizing radiation (IR) in not only accidental but also planned exposures (such as radiotherapy and interventional radiology). Deficiencies in DNA repair mechanisms would be involved on hypersensitivity to deterministic effects of IR. 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.

Under this context, individual radiosensitivity assessment was conducted in patients included in this research protocol that showed acute and/or late cutaneous reactions with grades 3 and 4 of the Toxicity Criteria of the Radiation Therapy Oncology Group and the European Organization for Research and Treatment of Cancer.

DNA repair capacity was evaluated through MN and comet assay for initial damage and after specific times of repair (0-120 minutes). DNA damage and repair capacity were quantified by the Olive tail moment.

Previous own studies have identified three subpopulations, characterized by the mean values of their repair mean half-time: healthy controls (2.6 ± 0.3 minutes), average-reactor cancer patients (4.7 ± 2.9 minutes) and over-reactor cancer patients (24.9 ± 10.4 minutes).

In this paper, 10 representative cases, in which the research protocol was applied, have been evaluated retrospectively. Therapeutic response and its correlation with radiosensitivity test results have also been studied.

Overall, 4 cases showed positive (favorable) local recovery and almost complete to complete remission of signs and symptoms after 5 to 12 months of the beginning of the treatment. In these patients, both MN frequencies and comet assay showed values compatible with normal (average) radiosensitivity. However, three cases showing average radiosensitivity presented complications attributed to radiation exposure (treatment or diagnosis) or to comorbidity factors. Finally, 3 cases presented a partial

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response with pain and acute repetitive crisis. In vitro radiosensitivity test results indicated that these patients (over-reactors) presented a greater risk than average patients of developing radiation toxicity.

These results would ensure in vitro radiosensitivity test to constitute one of the prognostic score parameters, jointly with applied radiation dose, radiation quality, localization (including thickness and vascularization of the dermis) in case of CRS for the design of therapeutic strategies. The appearance in delay of this syndrome often results in non specialized medical treatment, so the conformation of an ad hoc interdisciplinary group for CRS approach offer to these patients a content frame that provides them with the availability of diagnostic methods and therapeutic strategies, tailoring patient's treatment both in normal and in radiological emergency conditions.

KEYWORDS: Cutaneous Radiation Syndrome; In vitro Radiosensitivity Tests; DNA repair capacity

Patients 2005-2007	Pathology/diagnosis – therapeutic scheme	Clinical and <i>in vitro</i> radiosensitivity -T _{1/2} [min]			Clinical Follow-up	Patient and treatment related factors
		Healthy controls (2,6±0,3)	Average- reactor patients (4,7±2,9)	Over-reactor patients (24,9±10,4)		
G.Ce.	Breast Cancer/radiotherapy		5,76		Positive	
V.C.	Cardiopathy/interventional radiology Late Toxicity			19,8	Cicatrization Intense pain Microulcerations	Favorable location of injury
C.N.	Mixed tumor of the parotid gland /radiotherapy			14,8 ± 7,4	Positive Painless Cicatrization	
G.Ca.	Juvenile acne treatment with IR		7,3±1,8		Ischemia/reperfusion crisis Pruritus	Injury exposed to solar radiation
T.M.	Breast Cancer/radiotherapy			11,0 ± 4,7	Partial Response	Unfavorable location of injury
A.C.	Beta-therapy for surgical scar treatment		4,5±3,2		Ischemia/reperfusion necrosis Scar with positive response	
P.S. M.C.	Cardiopathy/interventional radiology Acute Toxicity		8,3±2,0		Partial Response Pain	Smoker Necrosis of flaps
F. J.L.	Thromboangiitis obliterans (Leo Buerger Disease)	3,0±1,0				
P. G.	Breast Cancer/radiotherapy		4,5±0,8		Partial Response	Unfavorable location of injury
M.Z.	Breast Cancer/radiotherapy	In progress			Partial Response	Unfavorable location of injury

Table 1: Correlation between clinical response and cellular radiosensitivity assessed by comet assay.

STUDY PROTOCOL FOR AN APPROACH BASED ON DIAGNOSIS AND THERAPY OF CUTANEOUS RADIATION INDUCED LESIONS

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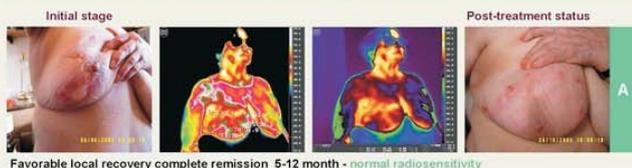
INTRODUCTION

The cutaneous radiation syndrome (CRS) constitute the most frequent accidental radiological event. It is caused by complex interactions between antiproliferative and proinflammatory process, following a clinically well-defined time pattern. There exist individual variations that could condition the response to ionizing radiation (IR) in not only accidental but also planned exposures. Deficiencies in DNA repair mechanisms would be involved on hypersensitivity to deterministic effects of IR.

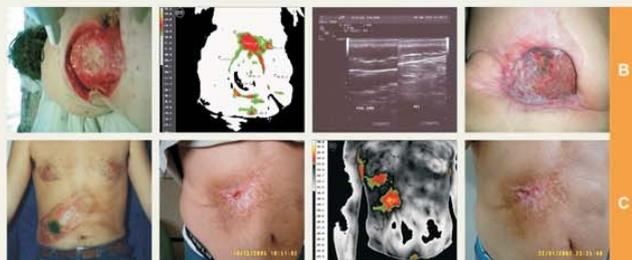
The "Hospital de Quemados del Gobierno de la Ciudad de Buenos Aires" (Burn Center) is one of the reference hospitals of the Medical Radiological Emergency Response Network of Argentina. In the frame of an agreement between the Burn Center and the Nuclear Regulatory Authority, a research project for diagnostic and therapeutic approach of CRS is in progress. Sixty seven persons, which developed acute and/or late CRS, were included in this protocol from 1997 to 2007, treated with an equivalent therapeutic scheme.

OBJECTIVE: To evaluate retrospectively the therapeutic response and its correlation with radiosensitivity test results in 10 representative cases, in which the research protocol was applied.

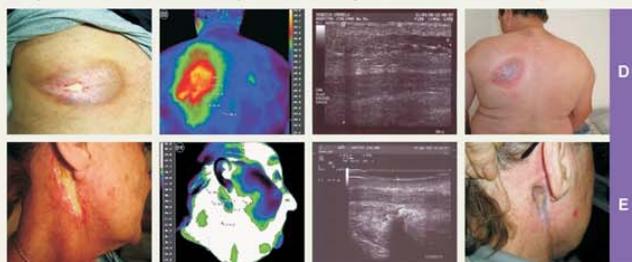
RESULTS AND DISCUSSION



Favorable local recovery complete remission 5-12 month - normal radiosensitivity



Complications attributed to radiation exposure + comorbidity factors - normal radiosensitivity



Partial response + pain+ acute repetitive crisis - over-reactors - hypersensitivity

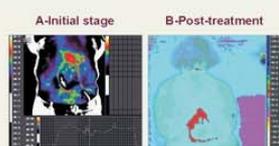
CRS PROGNOSTIC SCORE PARAMETERS for the design of therapeutic strategies

Based on:

- radiation dose
- radiation quality
- localization
- ✓ dermis thickness
- ✓ vascularization
- Extent, tissue involvement (in depth-damage)
- ✓ telethermography
- ✓ high frequency ultrasonography
- Individual radiosensitivity: in vitro radiosensitivity tests



Inflammation and ulceration of the frontal area correlating with a significantly higher skin temperature, in the telethermography Color Doppler ultrasound: left frontal area showing vascular obliteration

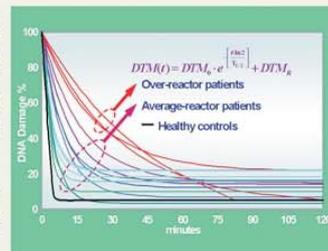


Telethermographies showing vascularization in the upper part of the lesion (A), evolving to reperfusion in its lower part, 8 month post-treatment (B)

MATERIALS AND METHODS

- In vitro individual radiosensitivity test applied: cytokinesis blocked micronucleus and alkaline single-cell microgel electrophoresis (comet) assays. Blood samples were irradiated in vitro with 2 Gy.

- DNA repair capacity was evaluated for initial damage and after specific times of repair. Captured images were analyzed by CASP image analysis software, quantified by the Olive tail moment and fitted by a mono-exponential model.



- DNA repair kinetic: Three subpopulations were identified, characterized by the means of their repair mean half-time.

- The therapeutic response was evaluated through clinical follow-up, serial photographic record and complementary tests (telethermography and high frequency ultrasonography). Therapeutic response and its correlation with radiosensitivity test results was studied.

- 2 cases showed favorable local recovery and partial to complete remission of signs and symptoms after 5 to 12 months of the beginning of the treatment. Both MN frequencies and comet assay showed values compatible with normal (average) radiosensitivity.
- 3 cases showing average radiosensitivity presented complications attributed to radiation exposure (treatment or diagnosis) or to comorbidity factors.
- 4 cases presented a partial response with pain and acute repetitive crisis. In vitro radiosensitivity test results indicated that these patients (over-reactors) presented a greater risk than average patients of developing radiation toxicity and were characterized as hypersensitivity patients.
- Heterotopic calcification was observed in 2 cases. This can be regarded as end-stage damage following high dose radiotherapy.
- Vulnerability to trauma of the affected areas was observed.

Cases	Clinical and in vitro radiosensitivity			Clinical follow-up	Patient and treatment related factors
	Healthy controls (2.6 ± 0.3)	Average-reactor patients (4.7 ± 2.9)	Over-reactor patients (24.9 ± 10.4)		
A Breast cancer /RT		5.76		Moist desquamation Favorable response	
Breast cancer /RT		4.5 ± 0.8		Partial response Flaps	
B Cardiopathy/IR		8.3 ± 2.0		Necrosis, pain, grafts Partial response	Smoker Necrosis of flaps
Juvenile acne treatment /RT		7.3 ± 1.8		Ischemia/reperfusion Crisis, pruritus	Injury exposed to solar radiation
C Betatherapy (scar treatment)		4.5 ± 3.2		Ischemia/reperfusion necrosis	High dose
D Cardiopathy/IR			19.8	Fibrosis, pain microulcerations	Favorable injury location
E Tumor of parotid gland/RT			14.8 ± 7.4	Fibrosis, ulceration, calcification, painless	Diabetes Hypothyroidism
Breast cancer /RT			11.0 ± 4.7	Fibrosis, ulcerations	
Ovary cancer /RT			18.4 ± 3.2	Ulceration, necrosis, calcifications	High dose
Thromboangiitis obliterans	3.0 ± 1.0				Leo Buerger disease

Correlation between clinical response and cellular radiosensitivity assessed by comet assay

CONCLUSIONS

Due to the appearance in delay of this syndrome, that often results in non specialized medical treatment, and the complexity of clinical manifestations of radiation disease in most patients, an interdisciplinary treatment at specialized centers make it necessary.

The conformation of such an ad hoc interdisciplinary group provides these patients with the availability of diagnostic methods and therapeutic strategies, tailoring patient's treatment both in normal and in radiological emergency conditions.

As lesions can be debilitating and life-threatening some of the patients included in this research protocol will require life-long therapy and follow-up.

Attributability of Health Effects at Low Radiation Doses

González, A.J.

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Buenos Aires, Argentina, 19-24 octubre 2008.
El presente trabajo se presentó oralmente durante el evento.

ATTRIBUTABILITY OF HEALTH EFFECTS AT LOW RADIATION DOSES

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Abstract

A controversy still persists on whether health effects can be alleged from radiation exposure situations involving low radiation doses (e.g. below the international dose limits for the public). Arguments have evolved around the validity of the dose-response representation that is internationally used for radiation protection purposes, namely the so-called linear-non-threshold (LNT) model. The debate has been masked by the intrinsic randomness of radiation interaction at the cellular level and also by gaps in the relevant scientific knowledge on the development and expression of health effects.

There has also been a vague use, abuse, and misuse of radiation-related risk concepts and quantities and their associated uncertainties. As a result, there is some ambiguity in the interpretation of the phenomena and a general lack of awareness of the implications for a number of risk-causation qualities, namely its attributes and characteristics.

In particular, the LNT model has been used not only for protection purposes but also for blindly attributing actual effects to specific exposure situations. The latter has been discouraged as being a misuse of the model, but the supposed incorrectness has not been clearly proven.

The paper will endeavour to demonstrate unambiguously the following thesis in relation to **health effects due to low radiation doses**:

- (i) **Their existence is highly plausible.** A number of epidemiological statistical assessments of sufficiently large exposed populations show that, under certain conditions, the prevalence of the effects increases with dose. From these assessments, it can be hypothesized that the occurrence of the effects at any dose, however small, appears decidedly worthy of belief. While strictly the evidence does not allow to conclude that a threshold dose level does not exist either. In fact, a formal quantitative uncertainty analysis, combining the different uncertain components of estimated radiation-related risk, with and without allowing for the uncertain possibility of a universal low-dose threshold, concludes that the evidence does not favour the existence of such a universal threshold. Consequently, radiation protection measures ought to be applied to radiation exposure situations involving low radiation doses.
- (ii) **They are improvable at individual level.** The effect occurrence on specific individuals is not demonstrable on a yes-no basis. Its reality is axiomatic: namely taken by granted as self-evident, solely based on the acceptance of the LNT hypothesis as the only true basis for argument or inference. It is unfeasible to demonstrate the existence of the effects by uncontested evidence: the truth, validity, or genuineness of their diagnosis for specific individuals cannot be tested and the diagnostic correctness cannot be checked.

- (iii) **Their individual causation is *counterfactual*.** The proposition ‘a radiation exposure situation caused a health effect on an individual’ cannot be explained in terms of the counterfactual conditional ‘if the radiation exposure situation had not occurred, then the health effect would not have occurred’.
- (iv) **Their occurrence is not individually *attestable*.** In addition to their improvability, any formal proof of the existence of a radiation health effect on any specific individual is generally absent and impossible to obtain at low radiation doses and cannot be established through scientific evidence.

The papers winds up that ***attributability***, namely the assumption that some health effect occurs as the result of a given low-dose radiation exposure situation, are distinct notions at the collective and individual level. It then concludes the following:

- Increases in the effect (collective) prevalence can be *attributable* in the sense that the radiological impact on a population can, under certain conditions, be ascribed, namely credited, assigned, and imputed to a specific exposure situation as its cause or source. *Attributability* is only conditional on the assumption that the relationship between the number of people being exposed and their doses is robust enough to make epidemiological attestability feasible (Strictly, the population would also need to be identical to those populations studied epidemiologically).
- Conversely, at the individual level, stochastic health effects at low doses are, at this time of biological understanding, unfeasible to be credited, assigned and imputed and consequently ascribed to a specific exposure situation;
- However, if *attributability* is taken to be a stochastic notion, then a conditional probability of causation can be theoretically assigned (following Bayes’ theorem and using available scientific information). This stochastic *attributability*, nevertheless, will not be attestable.
- Therefore, while individual health effects can under certain theoretical assumptions be stochastically attributable, they can not be subjected to an attestable *attributability*.
- As a result, presently individual health effects can not be deterministically attributable to radiation exposure situations delivering low radiation doses and, thus, they may not be deemed attributable in codified legal systems.

KEYWORDS: Attributability; Low Radiation Doses.

Radionuclide Activity Concentrations in Spas of Argentina

Gnoni, G.; Czerniczyniec, M. and Palacios, M.

RADIONUCLIDE ACTIVITY CONCENTRATIONS IN SPAS OF ARGENTINA

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Argentina

Abstract. Geothermal waters have been used on a large scale for bathing, drinking and medical purposes. These waters often have a very high mineral content because solubility increases with temperature. Ground waters are in close contact with soil and rocks containing radium. Once formed by decay from radium, radon gas (Rn-222) may diffuse through rocks pores and geological discontinuities and may dissolve in these waters. Radon and other natural radionuclides are transported to the surface where radon can easily diffuse into the atmosphere. Then it may be possible to find out significant radon levels at places like geothermal spas. In this work the most important natural radionuclide activity concentrations in different thermal spas of Argentina were measured to characterize waters and to evaluate the exposure of workers and members of the public. Three passive methods were used to measure radon in air. One of them is an screening method based on the radon adsorption on activated charcoal. The other two methods are time integrated ones, CR-39 or Makrofol tracks detectors, which can be exposed between two and three months. To characterize waters other natural radionuclides have been also measured. Uranium concentration was measured by fluorimetry. Ra-226 and Pb-210 measurements were performed by radiochemical methods and liquid scintillation. The results obtained were compared with the guidelines values recommended by WHO and EPA for drinking waters and, in the case of radon in air, the results were compared with values established by BSS-115. In order to assess worker doses, the higher value measured for radon in air and real scenario data were taken into account. Moreover, in situ dose rate measurements were also performed and then compared with natural background values. In relation with water characterization, almost all values obtained for the geothermal waters analyzed were below the corresponding guidance levels. Taking into account the highest value measured of radon in air, the maximum annual effective doses calculated resulted below the corresponding dose criteria for members of the public and in the order of it for workers. As values measured may be variable, it is suggested that the Thermal Spas be re-evaluated to determine its evolution over time.

KEYWORDS: *Radon gas, geothermal water, natural radionuclides, thermal spa.*

1. Introduction

Spring waters have been used on a large scale for bathing, drinking and for medical purposes for many thousand of years. The practice of using geothermal waters has considerably increased in the last years, becoming thermal spas recreation places and health resorts. The importance of these waters is linked to the high water flow, arising at temperatures to 70°C. In relation with the types of geothermal waters, there are three main kinds of them. The first type is water issuing from a hot spring heated by geothermal heat. Water percolates deeply into the crust and is heated as it comes into contact with hot rocks. The second one is the water arising from rain that percolates and mixes with pre-existing geothermal water. A third type is water in volcanic zones that is heated by contact with magma. If water erupts in a jet it is called geyser; if it reaches the surface as steam it is called a fumarole.

These waters also have variable amounts of certain dissolved substances, as solute solubility increases with increasing temperature. In some cases these substances are identified with beneficial health effects, and act as a coadjuvant agent together with the water temperature in arthrosis, rheumatism, arthritis, respiratory tract disorders, etc. Among these dissolved substances there are radioactive elements of natural origin. These waters are in close contact with soil and rocks containing radium. Radon gas (Rn-222) is formed by its decay from radium and once formed it may diffuse through rocks, pores and geological discontinuities and dissolve in these waters. Radon and natural radionuclides are transported to the surface and radon can

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easily diffuse into the atmosphere and may produce elevated radon concentrations [1]. Some authors attribute beneficial properties to radon gas and perform different therapies volatilizing it from radon enriched water. These therapies are used for various diseases such as hypertension, osteoarthritis, asthma and diabetes, though the mechanisms are still not fully clarified where the gas is [2].

The presence of these natural radionuclides can give place to enhanced levels of radiation as well. The main sources of exposure are the inhalation of radon and its decay products released from the water into indoor air, the exposure to ambient gamma-radiation and the ingestion of thermal water containing natural radionuclides. The high radon concentrations in spas is a source of risk mainly for the staff and to a lesser extent to the patients. For example the EU identified in its directive hydrotherapy as a professional activity during which workers are exposed to natural radiation sources [3].

In this work the most important natural radionuclide activity concentrations in different geothermal spas of Argentina were measured to characterize air and water and to evaluate the exposure of workers and members of the public. The first assessed spa has been chosen because of the volcanic origin of its geothermal water (Thermal Spa 1). It is being evaluated since 2005 to determine its evolution over time. In 2007 a group of nine spas that use water coming from aquifer systems (groundwater reservoirs) started to be monitored (Thermal Spa 2 to 10).

2. Materials and methods

2.2 Radon in air and equilibrium factor measurements

In order to characterize the air, radon detectors were placed at different points of each thermal spa. Once the exposure period has finished, the detectors were taken for analysis to the ARN laboratories. In the case of thermal spa 1, the spa and the town around it are constructed in a fumarole area. People who work in the spa, live in its neighbourhood. Due to these situations it is also important to measure radon in dwellings and other places nearby the spa.

Three passive methods were used to measure radon in air. One of them is a screening method with a maximum exposure time of seven days, based on the radon adsorption on activated charcoal and its subsequent measurement in a liquid scintillation counter [4]. The other two methods are time integrated ones that use CR-39 or Makrofol tracks detectors, and can be exposed between two and three months. The detector with CR-39 determines only the average radon concentration, while the other method measures both radon concentration and the equilibrium factor (F) between radon and its daughters, since it uses two Makrofol passive track detectors in the same device. [5-9].

2.3 Radionuclides in water measurements

In order to characterize waters, two samples were collected at each sampling point, one of them to measure the dissolved radon and the other one to determine natural uranium, Ra-226 and Pb-210. The water samples were collected at the source of the spring or as close as possible to this point. Also water samples from different baths used for treatments in spa 1 were taken. It can also be noticed that in thermal spa 1 there are different water sources to be evaluated. The water samples were taken for analysis to the ARN laboratories.

The radon determination was performed by liquid scintillation [10]. Uranium concentration was measured by fluorimetry or by KPA (Kinetic Phosphorescence Analysis) [11]. Ra-226 and Pb-210 determinations were performed by radiochemical methods based on their precipitation as sulfates, and final measurement by liquid scintillation [12].

2.4 Dose rate measurements

During the first survey, dose rate measurements were carried out at different sampling points of thermal spa 1. Background measurements were performed in the surroundings of the place. The instrument used was a Geiger- Müller detector AUTOMESS 2174.

3. Results

In tables 1 to 3 air radon concentration measurements and equilibrium factor determination in Thermal Spa 1 and surroundings are shown.

Table 1: [Rn-222] (Bq/m³) at Thermal Spa 1.

Description	April 2005	December-march 2006		
	Activated charcoal	Activated charcoal	CR-39	Makrofol
Health office	140 ± 15	86 ± 10	120 ± 25	-
Bath A 3	260 ± 30	1543 ± 170	-	-
Office bath A 3	490 ± 50	279 ± 30	373 ± 70	-
Corridor bath B 5	300 ± 30	155 ± 20	-	68 ± 15
Bath C 6	-	205 ± 20	414 ± 80	-
Corridor bath C 6	280 ± 30	305 ± 30	254 ± 50	-
Bath D 7	770 ± 80	117 ± 15	-	-
Bath D, sulphurous water 7	250 ± 30	861 ± 90	877 ± 160	-
Corridor bath D 7	710 ± 70	177 ± 20	-	113 ± 20
Bath E 8	-	854 ± 90	1755 ± 340	-
Corridor bath F 9	600 ± 60	535 ± 60	459 ± 90	-

Table 2: [Rn-222] (Bq/m³) in dwellings and other places nearby Thermal Spa 1.

Description	April 2005	December-march 2006		
	Activated charcoal	Activated charcoal	CR-39	Makrofol
House 1 - Children bedroom	814 ± 80	-	425 ± 85	-
House 1 - Main bedroom	1080 ± 110	95 ± 10	-	-
House 1 - Dining room	1258 ± 130	93 ± 9	-	125 ± 25
House 2	-	28 ± 3	198 ± 40	-
House 4	-	38 ± 4	125 ± 25	-
House 5	-	24 ± 3	-	12 ± 3
Police Station		958 ± 100	642 ± 120	
Hotel 1	44±6			
Hotel 2	29±5			
Restaurant	14±5			

Table 3: Measured equilibrium factor at Thermal Spa 1 and surroundings.

Description	December-march 2006
Corridor bath B	0.76
Corridor bath D 7	0.57
House 5	0.63

Dose rate measurements are shown in table 4. Background measurements were performed in the surroundings of the spa and were within natural radiation levels (0.1 – 0.2 µSv/h).

Table 4: Dose rate measurements at Thermal Spa 1 and surroundings.

Description	Dose rate ($\mu\text{Sv/h}$)
Water at spring 1	0.20
Water at spring 2	0.15
Water at spring 3	0.20
Surroundings of water spring 3	0.10
Outdoor bath 1	0.10
Outdoor bath 2	0.14
Ferrous water at spring	0.10
Sulphurous water at spring	0.10
Surroundings of sulphurous water at spring	0.11
Sulphurous water outdoor bath	0.20
Sulphurous water indoor bath	0.18
Health office	0.14
Corridor Health office	0.14
Bath E	0.10
Bath E Reception	0.20

In table 5, air radon concentration measurements in Thermal Spa 2 to 10 are shown.

Table 5: [Rn-222] (Bq/m^3) at Thermal Spas 2 to 10.

Thermal Spas	Description	CR-39 February–may 2008
	Jacuzzi	199
2	Emergency room	84
	Indoor swimming pool	87
3	Emergency room	35
	Emergency room	72
4	Women's locker room	45
	Medical office	21
5	Jacuzzi	29
6	Women's locker room	179
	Indoor/outdoor swimming pool	114
7	Outdoor swimming pool	107
	Dinning room	91
8	Men's locker room	82
	Indoor swimming pool	89
	Men's locker room	175
9	Jacuzzi	45
10	Emergency room	69

Radon, Natural uranium, Ra-226 and Pb-210 concentrations were also measured in different water samples as shown in table 6 to 8. Uranium determinations in april 2005 and april 2006 were performed by fluorimetry, and in the other surveys by KPA. A lower detection limit can be achieved with KPA.

Thermal Spa	Description	April 2005	April 2006	December 2006	April 2007	February 2008
Thermal Spa 1	Drinking water 1	1200 ± 700	-	< LD; LD=98	-	-
	Drinking water 2	1700 ± 800	-	< LD; LD=98	-	-
	Water at spring 1	4500 ± 1400	8700 ± 2000	-	-	6500 ± 1600
	Water at spring 2	3300 ± 1100	7000 ± 1700	-	-	15600 ± 2700
	Water at spring 3	1700 ± 700	5400 ± 1400	< LD; LD=98	-	5100 ± 1300
	North spring	5752 ± 1500	9000 ± 2000	-	-	-
	South spring	11083 ± 2400	4200 ± 1200	-	-	-
	Outdoor bath 1	-	28900 ± 6100	5163 ± 1084	-	19800 ± 4200
	Outdoor bath 2	-	29800 ± 6300	3591 ± 762	-	26000 ± 6000
	Ferrous water at spring	1900 ± 700	1100 ± 800	< LD; LD=98	-	-
	Ferrous water in bath	-	-	2636 ± 571	-	-
	Sulphurous water at spring	2300 ± 800	6100 ± 1500	862 ± 238	-	7400 ± 1800
	Sulphurous water outdoor bath	-	-	3327 ± 708	-	5500 ± 1400
	Sulphurous water indoor bath	-	-	2518 ± 574	-	9100 ± 1900
	Volcano water	-	10900 ± 2400	< LD; LD=98	-	1600 ± 800
Thermal Spa 2	Water at spring	-	-	-	<LD; LD=1114	<LD; LD=1072
Thermal Spa 3	Water at spring	-	-	-	<LD; LD=1114	<LD; LD=1072
Thermal Spa 4	Water at spring	-	-	-	5230 ± 1094	2168± 461
Thermal Spa 5	Water at spring	-	-	-	2132 ± 461	2036 ± 435
Thermal Spa 6	Water at spring	-	-	-	2563 ± 549	<LD; LD=1072
Thermal Spa 7	Water at spring	-	-	-	2625 ± 560	1835 ± 391
Thermal Spa 8	Water at spring	-	-	-	5957 ± 1246	5249± 1094
Thermal Spa 9	Water at spring	-	-	-	2273 ± 489	3388 ± 712
Thermal Spa 10	Water at spring	-	-	-	2869 ± 609	3113± 655

Table 6. [Rn-222] (Bq/m³) in geothermal waters used for medical purposes at Thermal Spas

Table 7. [Unat] [Ra-226] [Pb-210] in geothermal waters at Thermal Spa 1.

	Uranium			Ra-226			Pb-210		
	Total number of samples	Samples below detection limit (dl) ($\mu\text{g/l}$)	Values above detection limit ($\mu\text{g/l}$)	Total number of samples	Samples below detection limit (dl) (Bq/l)	Values above detection limit (Bq/l)	Total number of samples	Samples below detection limit (dl) (mBq/l)	Values above detection limit (mBq/l)
April 2005	11	11 (dl= 5)	-	-	-	-	-	-	-
April 2006	11	8 (dl= 8,2)	Mín: 13 \pm 2 Máx: 21 \pm 2	11	10 (dl=0.02)	0.03 \pm 13	11	11 (dl=48)	-
December 2006	5	4 (dl= 0,3)	0.5 \pm 0.2	5	5 (dl=0.04)	-	5	5 (dl=59)	-
February 2008	-	-	-	9	5 (dl=0.01)	Mín: 0.02 \pm 0.03 Máx: 0.04 \pm 0.02	9	9 (dl=59)	-

Table 8. [Unat] [Ra-226] [Pb-210] in geothermal waters at Thermal Spas 2 to 10.

	Uranium			Ra-226			Pb-210		
	Total number of samples	Samples below detection limit (dl) ($\mu\text{g/l}$)	Values above detection limit ($\mu\text{g/l}$)	Total number of samples	Samples below detection limit (dl) (Bq/l)	Values above detection limit (Bq/l)	Total number of samples	Samples below detection limit (dl) (mBq/l)	Values above detection limit (mBq/l)
April 2007	9	0 (dl= 0,3)	Mín: 0.52 \pm 0.05 Máx: 27.10 \pm 2.70	9	6 (dl=0.01)	Mín: 0.03 \pm 0.02 Máx: 0.10 \pm 0.02	9	7 (dl=59)	Mín: 72 \pm 27 Máx: 170 \pm 34
February 2008	9	0 (dl= 0,3)	Mín: 0.14 \pm 0.01 Máx: 28.41 \pm 2.80	9	3 (dl=0.01)	Mín: 0.05 \pm 0.03 Máx: 1.13 \pm 0.10	9	9 (dl=59)	-

4. Discussion and conclusions

In relation with water characterization, almost all values obtained for the geothermal waters analyzed are below the corresponding guidance levels.

The values recommended by World Health Organization (WHO) for drinking waters are 100.000 Bq/m³ for Rn-222, 100 mBq/l for Pb-210 and 1 Bq/l for Ra-226 [13].

For natural uranium, according to Argentinean regulations, the guidance level is 100 $\mu\text{g/l}$ from the point of view of its toxicology [14]. If we consider natural uranium from the radiological point of view, we can calculate the concentration in Bq/l (considering that its specific activity is 25,3 Bq/mg, recommended by international recommendations), and then obtain the effective dose taking into account the dosimetric factors of argentinian regulations [15]. For the highest value determined in these waters, 28.41 $\mu\text{g/l}$, we obtain 0.02 mSv/year. This value is below the 0.1 mSv/year that OMS takes into account to obtain the values recommended for natural radionuclides in water [13].

In the case of Pb-210, a value of 170 mBq/l was detected during april 2007, corresponding to Termal Spa 5. Although these waters are not used as drinking waters, in order to perform a conservative approach, the annual effective dose by ingestion can be calculated. Considering an annual ingestion of 730 liters and a dosimetric factor of $6,9 \cdot 10^{-4}$ mSv/Bq)[15], the resulting annual dose would be 0,08 mSv/a, a value far bellow the limit of the public (1 mSv/a)[15].

In the case of Ra-226, a value of 1.13 Bq/l was measured during february 2008, corresponding to Thermal Spa 2. Considering again an annual ingestion of 730 liters and a dosimetric factor of $6,9 \cdot 10^{-4}$ $2,8 \cdot 10^{-7}$ Sv/Bq) [15], the resulting annual dose would be 0,23 mSv/a.

Radon concentration measurements in air with activated charcoal are variable with time due to the short exposure period of this screening technique. It is a screening method that is used as a preliminary value. Therefore, to estimate the annual effective dose with more representative data, values obtained from CR-39 or Makrofol tracks detectors are used.

The highest radon concentration value in dwellings was measured in House 1. It was informed that this house was built in a fumarole area. In the case of the police station, the high value measured is concordant with the fact that this building is located in the same fumarole area than House 1.

An annual radon concentration of 400 Bq/m³ is the criteria for establishing intervention in a dwelling, according with argentine normative [15]. As people live there only six months per year because the rest of the year the place and sorroundings are covered with snow, the maximum dose that a member of the public may receive related with radon in air was calculated. Having into account this situation, it was assumed a spent time of 3500 hours per year. With the objective of assessing the dose, the highest radon concentration value measured in dwellings (425 Bq/m³) was considered. The annual effective dose calculated in this way is 3.7 mSv.y⁻¹. In order to assess the dose that a worker may receive, taking into account the highest value measured at the Thermal Spa (1755 Bq/m³), with a spent time of 1000 hours per year, the annual effective dose resulted in 5.6 mSv.y⁻¹. The dose criteria for establishing radon intervention levels, both for members of the public and for workers are 7 mSv.y⁻¹ [16]. The maximum annual effective doses calculated resulted below the corresponding dose criteria for members of the public and in the order of it for workers.

Finally, it is suggested that the Thermal Spas be re-evaluated to determine radionuclides evolution over time.

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A Management Tool for Improvement (Improving) IMS Stations Performance

Mohammad, D.; Pantin, A. and Quintana, E.

Presentado en: Workshop on Operation and Maintenance - CTBO
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El presente trabajo se presentó con poster y oralmente durante el evento.

A MANAGEMENT TOOL FOR IMPROVEMENT (IMPROVING) IMS STATIONS PERFORMANCE

Mohammad, D.; Pantin, A. and Quintana, E.

Argentina CTBT Project
Nuclear Regulatory Authority
Argentina

IMS-Maintenance is a software under development by the personnel of ARN-CTBT office, intended for the technical management of IMS stations under ARN's responsibility.

This software integrates the whole control of the equipment in an IMS station, including the management of consumables used in daily or periodic operation. It permits an easy retrieval (access) of the technical features, suppliers, etc. of each item of the equipment, as well as its maintenance history and scheduled task. Although this project is still in a starting phase, a preliminary version of this software is being used in radionuclide station RN01 (Buenos Aires), having already demonstrated its usefulness and potential for the improvement of IMS station management.



A MANAGEMENT TOOL FOR IMPROVING IMS STATIONS PERFORMANCE

Mohammad Diego - Pantin Andrés – Quintana Eduardo

Nuclear Regulatory Authority (ARN) - Argentina

INTRODUCTION

IMS-Maintenance is a software under development by the personnel of ARN-CTBT Project office, intended for the technical management of IMS stations under ARN's responsibility.

❖ RESPONSIBILITY OF THE ARN:

- In Argentina ARN is responsible for the installation and operation of the CTBTO radionuclide and infrasound stations and the certified laboratory.

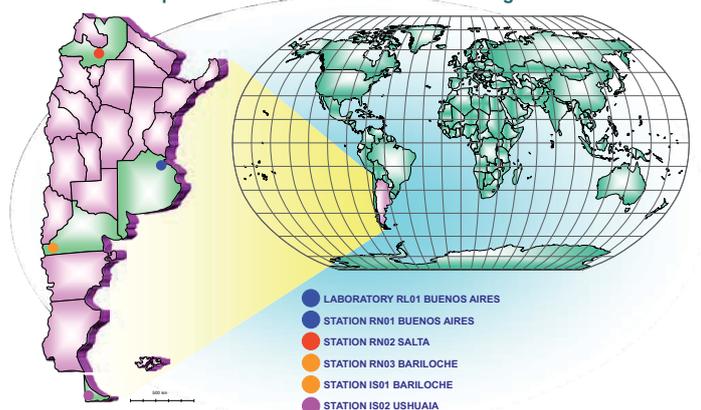
❖ MAIN OBJECTIVES OF THE SOFTWARE:

- Control and maintenance of the station equipment.
- To establish process standards.
- To reduce the number of outages of the equipment.

❖ GENERAL DESCRIPTION:

- Equipment maintenance administration software.
- It establishes uniform approaches for support tasks.
- It would allow to establish general norms of maintenance and to include them in the operation manual.
- Simple, easy and user friendly.
- Great administration capability.

The map shows facilities of the CTBT IMS in Argentina



ARP01



ARP03



IS02



Sampler Module ARP01



Measurement Room ARP01

❖ ADVANTAGES OF THE SOFTWARE:

- Innovative and effective software.
- Self-management.
- Dynamic and easy to use.
- Simplifies maintenance.

❖ TECHNOLOGY:

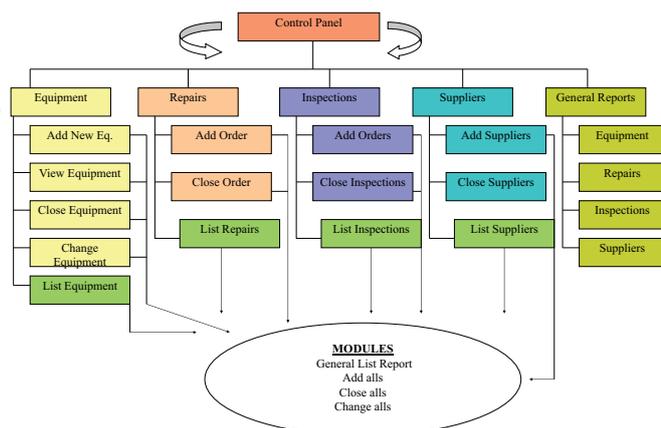
- Compatible with platforms:
 - Windows 98 / Millennium / XP
 - Windows NT / 2000
- Soon available for platforms:
 - Linux
 - Windows Vista
- Compatible databases:
 - ACCESS
 - SQL SERVER
- TCP IP compatible protocol

❖ CONTROL CAPACITY OF :

- General station equipment:
 - Power generator
 - Air Conditioner
 - Collection fans
 - Electric boards
 - Meteorological sensors
 - Detectors
 - Laboratory/Room sensors

❖ CAPABILITIES AND BENEFITS:

- Effective control of the periodic maintenance.
- Keeps reports of recurrent fails.
- Practical equipment history report for technicians.



❖ CURRENT STATE OF THE DEVELOPMENT:

- Chronogram of Development according to date :
 - User's interface improvement.
 - Equipment inventory.
 - Formulating the necessary processes for the operation.

CONCLUSION

This software integrates the whole control of the equipment in an IMS station, including the management of consumables used in daily or periodic operation. It permits an easy retrieval of the technical features, suppliers, etc. of each item of the equipment, as well as its maintenance history and schedules. Although this project is still in an initial phase, a preliminary version of this software is being used in radionuclide station RN01 (Buenos Aires), having already demonstrated its usefulness and potential for the improvement of IMS station management.

PARTE II

Resúmenes de publicaciones en revistas

AN ANALYTICAL CALCULATION OF THE PEAK EFFICIENCY FOR CYLINDRICAL SOURCES PERPENDICULAR TO THE DETECTOR AXIS IN GAMMA-RAY SPECTROMETRY *

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An analytical expression for the so-called full-energy peak efficiency $\epsilon(E)$ for cylindrical source with perpendicular axis to an HPGe detector is derived, using point-source measurements. The formula covers different measuring distances, matrix compositions, densities and gamma-ray energies; the only assumption is that the radioactivity is homogeneously distributed within the source. The term for the photon self-attenuation is included in the calculation. Measurements were made using three different sized cylindrical sources of ^{241}Am , ^{57}Co , ^{137}Cs , ^{54}Mn , and ^{60}Co with corresponding peaks of 59.5, 122, 662, 835, 1173, and 1332 keV, respectively, and one measurement of radioactive waste drum for 662, 1173, and 1332 keV.

* Publicado en: Applied Radiation and Isotopes; vol.66, no.8, p.1123-1127, 2008

A GENETIC ALGORITHM APPROACH TO ROUTINE GAMMA SPECTRA ANALYSIS *

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In this work we present an alternative method for performing routine gamma spectra analysis based on genetic algorithm techniques. The main idea is to search for patterns of single nuclide spectra obtained by simulation in a sample spectrum targeted for analysis. We show how this approach is applied to the analysis of simulated and real target spectra, and also to the study of interference resolution.

* Publicado en: Journal of Instrumentation; vol.3, no.1, 2008

NATURAL RADIONUCLIDE ACTIVITY CONCENTRATIONS IN SPAS OF ARGENTINA *

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Geothermal waters have been used on a large scale for bathing, drinking and medical purposes. These waters can contain natural radionuclides that may increase the exposure to people. In this work the most important natural radionuclide activity concentrations in different thermal spas of Argentina were measured to characterize waters and to evaluate the exposure of workers and members of the public.

* Publicado en: AIP Conference Proceedings; vol.1034, no.1, p. 242-245, 2008

ANALYSIS OF THE MAIN THERMOLUMINESCENT PEAK OF THE GLOW CURVE OF $K_2 YF_5:Pr^{3+}$ CRYSTALS EMPLOYING A MODEL OF INTERACTIVE TRAPS *

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By employing a model of interactive traps which conforms to experimental findings the glow curve of $K_2 YF_5 : Pr^{3+}$ compounds has been analysed. A novel algorithm, which allows the decoupling of the equations describing the carrier traffic among traps, recombination centres and energy bands, is reported. An important conclusion drawn from the results is that it is not always correct to think of each single peak as related only to one trap.

* Publicado en: Radiation Measurements; vol. 43, no. 2-6, p. 208-212, 2008

^{40}K , ^{115}Cs and ^{226}Ra AND PLANT CONTENT IN SEMINATURAL GRASSLANDS OF CENTRAL ARGENTINA *

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Activity concentrations of ^{40}K , ^{226}Ra and ^{137}Cs have been analyzed in soil and plant samples, collected in permanent grassland in central Argentina. Two near areas (A1 and A2) under field conditions with soil undisturbed at least in the last four decades were selected. For each of the three studied radionuclides we do not find differences in the inventories between both areas. The inventories range from 143 kBq m⁻² to 197 kBq m⁻² for ^{40}K and from 13 kBq m⁻² to 18 kBq m⁻² for ^{226}Ra . The vertical distributions of ^{40}K and ^{226}Ra are uniform through de soil profile. For ^{137}Cs the inventories range from 0.33 kBq m⁻² to 0.73 kBq m⁻². In spite of ^{137}Cs inventories are similar in both areas the distribution through vertical profile is different. ^{137}Cs activity concentration has a maximum for layers 5-10 cm depth in A1 and 10-15 cm depth in A2. For deeper layers both areas show similar activity concentrations. The diffusion coefficient (D_s) and convection velocity (v_s) are estimated with a convection-diffusion model. D_s values are in the range reported in the bibliography, while v_s values are one order of magnitude higher. After 40 years most ^{137}Cs fallout is still in the layer 10-15 cm depth. The great penetration of ^{137}Cs (25 cm) in these soils may be the result of a high sand and low fine materials content. ^{137}Cs and ^{226}Ra were not detected in grass samples. Activity concentration of ^{40}K in vegetal samples ranges from 116 Bq kg⁻¹ to 613 Bq kg⁻¹. The TF values obtained for ^{40}K show a lognormal distribution and ranges from 0.05 to 0.42.

* Publicado en: AIP Conference Proceedings; vol.1034, no.1, p. 273-276, 2008

PARTE III

Informe Nacional para la Convención de Seguridad Nuclear

National Nuclear Safety Report. Presentation

Compilación a cargo de:
Waldman, R. M.; Navarro, N.R. and Calvo, J.C.

Presentado en: 4th Review Meeting on the Convention on Nuclear Safety.
Viena, Austria, 14-25 abril 2008

(El Informe presentado en la citada reunión es el editado en el año 2007 (Fourth National Nuclear Safety Report) y su versión completa puede ser consultada en el sitio web de la ARN: www.arn.gob.ar)

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