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Propiedades electrónicas del Be y Al mediante la técnica de dispersión Compton

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Propiedades electrónicas del Be y Al mediante la técnica de dispersión Compton

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Resumen

En este trabajo se presentan los resultados obtenidos en el estudio de berilio y aluminio utilizando la técnica de dispersión Compton para la determinación de las propiedades electrónicas de materiales. El método se basa en la irradiación de muestras utilizando un haz delgado de fotones mono-energéticos de 59,54 keV producto del decaimiento radiactivo del Am-241. La radiación dispersa es colectada mediante un detector semiconductor de alta resolución de germanio hiperpuro posicionado a un ángulo de 90°. El espectro medido es comúnmente llamado perfil Compton y contiene información de interés sobre la estructura electrónica del material.

Los resultados experimentales se comparan con cálculos teóricos realizados con la Teoría del Densidad Funcional observándose una buena concordancia. Sin embargo estos resultados presentan discrepancias con la mayoría de las bibliotecas utilizados en códigos de simulación por Monte Carlo basadas en los valores tabulados por Biggs, Mendelsohn y Mann en 1975 sobreestimándose así la radiación dispersa sobre un material.

Introducción

En una colisión fotón-electrón solo una fracción de energía es transferida al electrón que es eyectado de la órbita produciendo un nuevo fotón de menor energía llamado dispersión incoherente.

La medición de un espectro de dispersión recibe el nombre de perfil Compton y permite obtener información sobre la estructura electrónica del medio material sean estos sólidos, líquidos o gases [1,2].

La contribución de un orbital atómico nl al perfil Compton viene dado por

$$J_{nl}(q) = \frac{1}{2}\int_q^{\infty} |\chi_{nl}(p)|^2 p\ dp = \frac{1}{2}\int_q^{\infty} \rho_{nl}(p)^2 p\ dp$$

Aquí $\chi_{nl}(p)$ representan la función de onda en el espacio de momento y $\rho_{nl}(p)$ la densidad electrónica.

En estas condiciones el perfil Compton $J(q)$ debe cumplir lo siguiente

$$\int_{-q}^{+q} J_{nl}(q)\ dq = Z$$

donde Z es el número atómico del elemento y siendo $q(E)$ la proyección del momento transferido k sobre el momento del electrón $\boldsymbol{p}$ luego de la colisión.

$$q(p) = -\frac{\boldsymbol{k}.\boldsymbol{p}}{k}$$

Expresando esto último en unidades de energía se obtiene:

$$q(E) = -137\frac{E_i - E_f + E_iE_f(1-\cos\theta)/m_0c^2}{({E_i}^2 + {E_f}^2 - 2E_iE_f\cos\theta)^{1/2}}$$

En esta expresión anterior $m_0c^2 = 511{,}0034$ keV, E_i y E_f representan la energía del fotón incidente y disperso respectivamente.

$\chi(p)$ es una función calculada a partir de la transformada de Fourier de la función de onda $\psi_{nl}(r)$ o bien utilizando la función esférica de Bessel de primer orden

$$\chi_{nl}(p) = \frac{2}{\pi}\int_0^{\infty} R_{nl}(r) j_l(pr) r^2\ dr$$

donde $R_{nl}(r)$ es la función de onda radial que puede calcularse a partir del método de Hartree-Fock para orbitales internos, o bien haciendo utilizando pseudos orbitales atómicos haciendo uso de la teoría del funcional densidad (DFT).

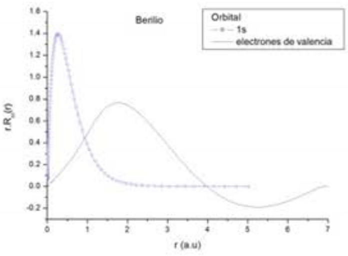

Fig. 1

En la figura 1 se muestra las funciones de onda calculadas por DFT para un cristal *hexagonal* de berilio (P63/mmc) en la notación de Hermann-Mauguin.

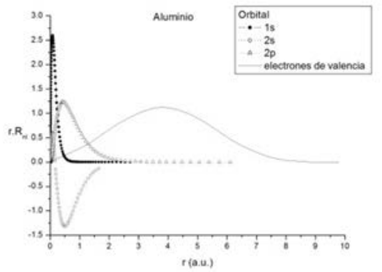

Fig. 2

La figura 2 muestra las funciones de onda de un cristal *fcc* (Fm-3m) de aluminio.

Procedimiento experimental

En este trabajo se presentan algunos de los resultados obtenidos en metales de berilio y aluminio con purezas superiores a 99,8 %. El espectro fue colectado a un ángulo 90°, utilizando un detector semiconductor de alta resolución de germanio hiperpuro [3].

Se ha utilizado un radioisótopo emisor gamma (59,54 keV) producto del decaimiento radiactivo del Am-241 como fuente mono-energética de fotones.

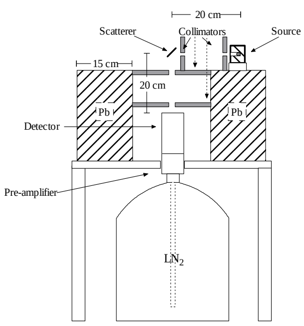

Fig. 3

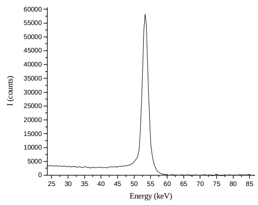

Fig. 4

En las figuras 3 y 4 se esquematiza el montaje experimental y un espectro colectado.

Resultados

Valiéndose de los resultados obtenidos con la teoría del funcional densidad presentados en las figuras 1 y 2 fue posible determinar las distintas funciones de $\chi_{nl}(p)$ y así determinar los valores de $J_{total}(q)$.

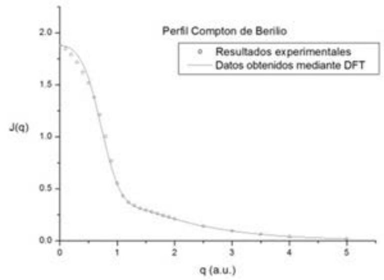

Fig.5

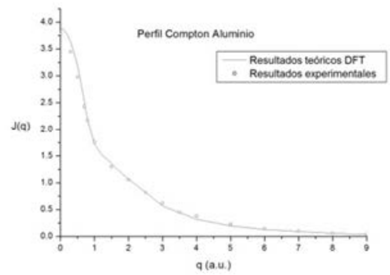

Fig.6

Conclusiones

En este trabajo se han presentados resultados teóricos basados en la teoría del funcional densidad para el cálculo del perfil Compton y comparados experimentalmente a partir del espectro de dispersión colectado.

Los resultados obtenidos hasta el momento tienen vital importancia tanto en física del sólido como en física de la radiación. Se resalta esto último ya que existen numerosos programas basados en el método de Monte Carlo que utilizan librerías de perfiles Compton para el cálculo de radiación dispersa y que lamentablemente no se condicen con los valores experimentalmente observados sobreestimando en todos los casos la radiación dispersa sobre distintos materiales.

Referencias

[1] M.J. Cooper, et. al., *X-ray Compton Scattering*, Oxford University Press, New York, 2004.

[2] B.G. Williams (Ed.), *Compton Scattering*, McGraw-Hill, New York, 1977.

[3] J. Aguiar, H. O. Di Rocco and A. Arazi, Physica B 406, 354 (2011).
[4] F. Biggs, L. Mendelsohn and J. Mann. Hartree Fock Compton profiles for elements. Atomic Data and Nuclear Data Tables 16, 3 (1975)

Marco legal del acceso a la información y participación ciudadana en el ámbito de la actividad nuclear

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MARCO LEGAL DEL ACCESO A LA INFORMACIÓN Y PARTICIPACIÓN CIUDADANA EN EL ÁMBITO DE LA ACTIVIDAD NUCLEAR

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I.- Introducción

El derecho de acceso a la información por parte de los ciudadanos sobre las actividades vinculadas al desarrollo científico – tecnológico de la energía nuclear con fines pacíficos fue evolucionando en el tiempo. Los gobiernos comenzaron a percibir la necesidad y los beneficios de informar a la comunidad, que manifestaba ciertos prejuicios respecto de la actividad nuclear como consecuencia del lanzamiento de las bombas nucleares de Nagasaki e Hiroshima.

Con el advenimiento del derecho ambiental y la influencia de sus principios, se fue imponiendo en el ámbito nuclear la idea de la transparencia en la información y la importancia de que tanto los habitantes de los países con desarrollos nucleares como de los países vecinos que pudieran verse afectados por los efectos transfronterizos de las radiaciones ionizantes, tuvieran acceso a la información y pudieran participar activamente.

El acceso a la información y la participación ciudadana ha sido institucionalizado y plasmado en la normativa internacional a través de convenciones internacionales suscriptas por nuestro país y en el ámbito nacional a través de la Constitución Nacional, las Constituciones provinciales, la Constitución de la Ciudad de Buenos Aires, las Leyes N° 25.675, 25.831 y en el Decreto PEN N°1172/03, entre otras.

El presente trabajo tiene por objeto hacer un repaso sobre el marco legal relacionado con el acceso a la información en la actividad nuclear.

II.- Derecho de Acceso a la Información y Participación Ciudadana

Comenzaremos el presente trabajo clarificando los conceptos de derecho de acceso a la información y derecho a la participación ciudadana.

Se entiende por derecho de acceso a la información, a *"la prerrogativa que tiene toda persona de solicitar y obtener, en tiempo y forma adecuada, información que sea considerada de carácter público y se encuentre en poder del Estado"*[1].

[1] "Derecho de Libre Acceso a la Información Pública"- Nápoli, A. M. y Vezzulla, J. M.

Corresponde hacer una distinción entre el derecho a la información y el derecho de acceso a dicha información. El derecho a la información implica que el Estado se encuentra obligado a la producción, elaboración y difusión de información, mientras que el derecho de acceso a la información pública, consiste en la posibilidad que tiene todo ciudadano de acceder a la información administrada por el Estado.

El Estado para garantizar efectivamente este derecho debe organizar la información, procesarla, clasificarla y establecer un sistema que permita el acceso y la selección de la información requerida.

Por otro lado, la participación ciudadana refiere a la posibilidad que los ciudadanos se involucren en los asuntos públicos. Para que la participación sea posible es necesario un sistema de información sincero, objetivo y transparente y que esté al alcance de los ciudadanos, de modo que invite a la discusión pública[2].

III.-El Derecho de Acceso a la Información y Participación Ciudadana en el ámbito internacional

En la segunda mitad del siglo veinte, comenzó a ser motivo de preocupación pública el daño ambiental causado por la actividad humana. Con el advenimiento del derecho ambiental se enmarcó legalmente la responsabilidad por los daños ambientales causados por diferentes industrias, así como el reconocimiento de los derechos de acceso a la información ambiental y a la participación de los ciudadanos en la toma de decisiones que afectaran el medio ambiente.

En este contexto, en el año 1972 la Conferencia de las Naciones Unidas sobre el Medio Humano[3], formuló la Declaración de Estocolmo, reconocida como el primer documento internacional que establece expresamente el derecho a un ambiente sano y promueve la adopción de normas, a nivel nacional como internacional, que prevean el derecho a la información por parte de los ciudadanos.

La citada declaración en su principio Nº 19 establece:

"Es indispensable una labor de educación en cuestiones ambientales, dirigida tanto a las generaciones jóvenes como a los adultos y que presente la debida atención al sector de población menos privilegiado, para ensanchar las bases de una opinión pública bien informada y de una conducta de los individuos, de las empresas y de las colectividades inspirada en el sentido de su responsabilidad en cuanto a la protección y mejoramiento del medio en toda su dimensión humana. Es también esencial que los medios de comunicación de masas eviten contribuir al deterioro del medio humano y difundan, por el contrario, información de carácter educativo sobre la necesidad de protegerlo y mejorarlo, a fin de que el hombre pueda desarrollarse en todos los aspectos."

En el mismo sentido, la Declaración de Río sobre el Medio Ambiente y el Desarrollo[4], del año 1992, proclama en su principio Nº 10:

"El mejor modo de tratar las cuestiones ambientales es con la participación de todos los ciudadanos interesados, en el nivel que corresponda. En el plano nacional, toda persona deberá tener acceso adecuado a la información sobre el medio ambiente de que dispongan las autoridades públicas, incluida la información sobre los materiales y

[2] "Participación y Medio Ambiente"- Diez Arregi, P.
[3] http://www.pnuma.org/docamb/mh1972.php
[4] http://www.pnuma.org/docamb/dr1992.php

las actividades que encierran peligro en sus comunidades, así como la oportunidad de participar en los procesos de adopción de decisiones. Los Estados deberán facilitar y fomentar la sensibilización y la participación de la población poniendo la información a disposición de todos. Deberá proporcionarse acceso efectivo a los procedimientos judiciales y administrativos, entre éstos el resarcimiento de daños y los recursos pertinentes".

Por su parte, en la Declaración de Johannesburgo sobre el Desarrollo Sostenible[5], del año 2002, se reconoce que el desarrollo sostenible exige una perspectiva a largo plazo y una amplia participación en la formulación de políticas, la adopción de decisiones y la ejecución de actividades en todos los niveles.

Estas declaraciones erigen a la participación pública y al acceso a la información como herramientas imprescindibles para avanzar en el camino del desarrollo sostenible, y destacan la necesidad de que las personas, los grupos y las organizaciones participen en los procedimientos de evaluación de impacto ambiental, conozcan el mecanismo de adopción de decisiones y tomen intervención en él, en especial cuando esas decisiones pudieran afectar el medio en que viven.

Por otra parte a nivel regional, con la suscripción del Convenio sobre Acceso a la Información, Participación del Público en la Toma de Decisiones y Acceso a la Justicia en Materia de Medio Ambiente (Convenio de Aarhus)[6], la Unión Europea pretende sensibilizar e implicar a los ciudadanos en las cuestiones medioambientales y mejorar la aplicación de la normativa medioambiental y establece como contrapartida la obligación de los organismos gubernamentales a tener a disposición del público la información ambiental, sin necesidad de que los requirentes acrediten un interés particular en la misma.

En el mismo sentido, la Convención sobre Evaluación de Impacto Ambiental en un Contexto Transfronterizo (Convención de ESPOO)[7] establece en su artículo tercero que siempre que exista la probabilidad de que una actividad cause un impacto transfronterizo de carácter perjudicial y de magnitud apreciable, el país donde la actividad se llevará a cabo, cursará una notificación a todo país parte de la Convención que considere pueda ser afectado, a fin de que se celebren consultas apropiadas y efectivas.

En Latinoamérica ha sido suscripto el Acuerdo Marco sobre Medio Ambiente del Mercosur[8], que tiene por objeto el desarrollo sustentable y la protección del medio ambiente, mediante la articulación de las dimensiones económicas, sociales y ambientales, contribuyendo a una mejor calidad del ambiente y de la vida de la población.

El citado Acuerdo, establece en su artículo tercero que los Estados Partes deben promover una efectiva participación de la sociedad civil en el tratamiento de las cuestiones ambientales.

Asimismo, en su artículo séptimo dispone que los Estados Partes acordarán pautas de trabajo que contemplen, entre otras áreas temáticas, la educación, información y comunicación ambiental.

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[5] http://www.un.org/esa/sustdev/documents/WSSD_POI_PD/Spanish/WSSDsp_PD.htm

[6] Firmado en Dinamarca el 25/6/98.

[7] Firmado en Finlandia el 25/2/91.

[8] Suscripto en Paraguay. Aprobado por Ley N° 25.841. Publicada en el B.O. N° 30.318 de fecha 15/1/04.

Por otra parte, en el ámbito de la Gestión Pública, ha sido suscripta la Carta Iberoamericana de Participación Ciudadana[9], que reconoce que el derecho de acceso a la información pública sustenta un adecuado funcionamiento de la democracia puesto que es condición para garantizar la participación ciudadana en la gestión pública. Prevé la protección jurídica del acceso a la información y la necesidad de que cualquier negativa a brindar información se encuentre expresamente prevista en el ordenamiento. Al mismo tiempo, puntualiza diversos aspectos que deben garantizarse para hacer efectivo el derecho, entre otros la obligación de asistir al ciudadano en la búsqueda de información, la debida motivación de las decisiones que denieguen total o parcialmente el acceso y la falta de necesidad de declarar un interés determinado en la información requerida.

IV.- Derecho de Acceso a la Información Pública y a la Información Pública Ambiental

Como mencionamos precedentemente, el derecho de acceso a la información es una instancia de participación ciudadana por la cual toda persona ejercita su derecho a requerir, consultar y recibir información de los sujetos que tienen la obligación de brindar la misma, en virtud de la actividad que desarrollan[10] .

Más allá de su valor propio, la información es de vital importancia como medio para el ejercicio de otros derechos consagrados en nuestra Constitución Nacional. Así, se considera que *"...el derecho de Acceso a la Información Pública es un prerrequisito de la participación que permite controlar la corrupción, optimizar la eficiencia de las instancias gubernamentales y mejorar la calidad de vida de las personas al darle a éstas la posibilidad de conocer los contenidos de las decisiones que se toman día a día para ayudar a definir y sustentar los propósitos para una mejor comunidad"*[11].

El reconocimiento jurídico del derecho de acceso a la información, comenzó a vislumbrarse en un fallo de la Corte Suprema de Justicia de la Nación del año 1991, en el cual la Corte, al momento de expedirse sobre la procedencia de un resarcimiento por la responsabilidad originada en la publicidad de una noticia considerada inexacta, expresó que la Constitución Nacional en sus artículos 14 y 32 y el Pacto de San José de Costa Rica aprobado por Ley N° 23.054, contemplan el derecho de información, derecho de naturaleza individual, y agregó que este derecho se encuentra en conexión con el derecho a la información, derecho de naturaleza social, mediante el cual se garantiza a cada persona el conocimiento y la participación en todo cuanto se relaciona con los procesos políticos, gubernamentales y administrativos, los recursos de la cultura y las manifestaciones del espíritu como un derecho humano esencial[12] .

Previo a la reforma constitucional del año 1994, nuestro sistema de acceso a la información se encontraba justificado en el principio de publicidad de los actos de gobierno, propio del sistema republicano de gobierno adoptado por nuestro país.

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[9] Aprobada por la XI Conferencia Iberoamericana de Ministros de Administración Pública y Reforma del Estado. Portugal 25 y 26 de junio de 2009. Adoptada por la XIX Cumbre Iberoamericana de Jefes de Estado y de Gobierno. Portugal 30 de noviembre y 1 de diciembre de 2009.

[10] Artículo 3° del Decreto N° 1172/03- Publicado en el B.O. N° 30.291 de fecha 4/12/03.

[11] Considerando 7° del Decreto PEN N° 1172/03.

[12] "Vago, Jorge A. c/ Ediciones La Urraca y otros s/ daños y perjuicios"- CSJN 19/11/1991.

Con la reforma operada en el año 1994 se incorpora a nuestra Constitución Nacional, en su primera parte, el capítulo segundo "Nuevos derechos y garantías", estableciendo el "derecho al ambiente" y el correspondiente "deber de preservarlo", con la finalidad de procurar un desarrollo sustentable para las actuales y futuras generaciones –artículo 41 de la Constitución Nacional-.

Asimismo, el referido artículo 41 establece la obligación de las autoridades de proveer información y educación ambiental. De esta manera, si bien no resulta expresamente de la citada norma el derecho de solicitar y recibir información por parte de los ciudadanos, implícitamente surgen para el Estado las siguientes obligaciones: almacenar la información, hacerla sistemática y continua, ordenarla, seleccionarla a fin de facilitar el acceso a la misma.

En consecuencia, si bien la Constitución Nacional garantiza la publicidad de los actos de gobierno -artículos 1°, 33, 41, 42-, debemos acudir a fuentes externas (tratados internacionales) para encontrar el reconocimiento expreso del derecho de acceso a la información pública, con excepción de las siguientes normas:

- El Reglamento General de Acceso a la Información Pública para el Poder Ejecutivo Nacional, aprobado por el Decreto PEN N° 1172/03[13].

- El Régimen de Libre Acceso a la Información Ambiental en poder del Estado, establecido mediante Ley N° 25.831[14]

Con relación al Reglamento General de Acceso a la Información Pública para el Poder Ejecutivo Nacional –Decreto PEN N° 1172/03- cabe mencionar que, a través de éste, se institucionalizaron una serie de instrumentos tendientes a generar una nueva cultura orientada a mejorar la calidad de la democracia, garantizando así el flujo informativo entre los actores sociales y sus autoridades, a fin de asegurar el ejercicio responsable del poder.

Si bien el referido Decreto es de carácter restringido dado que solo abarca el derecho de peticionar a los organismos que funcionen bajo la órbita del Poder Ejecutivo, considerando el fin que implica el acceso a la información pública, constituye una herramienta de gran importancia.

Por otro lado, mediante la Ley N° 25.831 -Régimen de Acceso a la Información Pública Ambiental-, se establecieron las bases mínimas de aplicación a todo el territorio nacional, incluidas las provincias y la Ciudad Autónoma de Buenos Aires.

La referida ley, en su artículo segundo establece el concepto de información ambiental, expresando: "*Se entiende por información ambiental toda aquella información en cualquier forma de expresión o soporte relacionada con el ambiente, los recursos naturales o culturales y el desarrollo sustentable*".

Con relación al acceso a dicha información, en su artículo tercero establece lo siguiente: "*El acceso a la información ambiental será libre y gratuito para toda persona física o jurídica, a excepción de aquellos gastos vinculados con los recursos utilizados para la entrega de la información solicitada. Para acceder a la información ambiental no será necesario acreditar razones ni interés determinado. Se deberá presentar formal solicitud ante quien corresponda, debiendo constar en la misma la información requerida y la identificación del o los solicitantes residentes en el país,*

[13] Decreto PEN N° 1172/03. Publicado en el BO N° 30.291 de fecha 4/12/2003.
[14] Ley N° 25.831. Publicada en el BO N° 30.312 de fecha 7/04/2004.

salvo acuerdos con países u organismos internacionales sobre la base de la reciprocidad.

En ningún caso el monto que se establezca para solventar los gastos vinculados con los recursos utilizados para la entrega de la información solicitada podrá implicar menoscabo alguno al ejercicio del derecho conferido por esta ley."

El citado artículo establece como principio la gratuidad en el acceso a la información, e indica expresamente que no es necesario acreditar razones ni interés determinado para acceder a la misma.

Cabe señalar, que el artículo séptimo establece que la información ambiental solicitada puede ser denegada, rechazando su pedido total o parcialmente cuando pudiera afectarse la defensa nacional, la seguridad interior o las relaciones internacionales, cuando se encuentre sujeta a consideración de autoridades judiciales y su divulgación o uso pueda causar perjuicio al desarrollo del procedimiento judicial y cuando pudiera afectarse el secreto comercial o industrial o la propiedad intelectual o la confidencialidad de datos personales.

V.- El Derecho de Acceso a la Información Pública en el ámbito local

En el ámbito local, existe diversa normativa relacionada con el acceso a la información pública.

En la Constitución de la Provincia de Buenos Aires[15] el artículo 12 de la misma, reconoce el derecho a la información y a la comunicación del cual gozan todas las personas, y específicamente en el artículo 28 se consagra el derecho a la información en materia ambiental.

En dicho artículo, se establece que la Provincia en materia ecológica *"Deberá garantizar el derecho a solicitar y recibir adecuada información."*

La Constitución de la Ciudad Autónoma de Buenos Aires[16], en su capítulo cuarto, relativo al ambiente, establece: *"Toda persona tiene derecho, a su solo pedido, a recibir información sobre el impacto que causan o puedan causar sobre el ambiente, actividades públicas o privadas"*.

La Ley N° 104 de la Ciudad Autónoma de Buenos Aires[17], de Acceso a la Información, comprende a los órganos de la administración pública, empresas con participación estatal y a los poderes legislativo y judicial en su faz administrativa.

Asimismo, también en el ámbito de la Ciudad Autónoma de Buenos Aires, podemos encontrar la Ley N° 303 "Ley de Información Ambiental"[18] que considera información ambiental a cualquier tipo de investigación, informe, datos sobre el estado del ambiente y de los recursos naturales, las declaraciones de impacto ambiental, los planes y programas, públicos o privados.

En las constituciones de otras provincias como Córdoba, Formosa, La Rioja, San Juan, Salta, Jujuy y Tierra del Fuego, se garantiza el derecho al acceso a las fuentes de información pública.

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[15] http://www.gob.gba.gov.ar/dijl/constitucion.php

[16] http://www.buenosaires.gov.ar/areas/com_social/constitucion/constitucion.php?menu_id=11172

[17] Ley N° 104. Publicada en el BOCBA N° 600 de fecha 29/12/1998.

[18] Ley N° 303. Publicada en el BOCBA N° 858 de fecha 13/01/2000.

La Provincia de Jujuy posee la Ley N° 4.444[19], la cual se trata de una ley expresa de libre acceso a la información, reglamentándose de esta manera la publicidad de los actos de gobierno y el libre acceso a las fuentes oficiales de información de acuerdo a lo establecido en la Constitución Provincial.

Asimismo, la Provincia de Misiones cuenta con una ley que regula la información en materia ambiental, es la Ley N° 4.184 "Información Ambiental"[20].

Por otro lado, también cuentan con leyes sobre acceso a la información, las Provincias de Chubut (Ley N° 3.764)[21], de Córdoba (Ley N° 8.803)[22] y la Provincia de Tierra del Fuego (Ley N° 653)[23].

Por último, podemos citar a la Provincia de Corrientes, que cuenta con la Ley N° 5.533 "Ley de Información Ambiental"[24]

VI.- Sistema de Información Pública Ambiental

Cabe destacar que en el ámbito nacional funciona el Sistema de Información Ambiental Nacional (SIAN), el cual fue creado en el año 1998 mediante Decreto N° 146/98, de acuerdo a lo establecido en el artículo 41 de la Constitución Nacional, que define la responsabilidad de las autoridades con relación a la provisión de información ambiental.

Se trata de un sistema de representación federal, actualmente integrado por 24 nodos correspondientes a los organismos gubernamentales ambientales de cada provincia y al de nivel nacional (la Secretaría de Ambiente y Desarrollo Sustentable) y 6 nodos de otras instituciones vinculadas a la temática ambiental[25].

Sus objetivos son: recopilar y procesar información ambiental con el fin de ponerla a disposición de los organismos gubernamentales ambientales, no gubernamentales y la comunidad; proveer al sector gubernamental instrumentos que faciliten los procesos de toma de decisiones en materia de gestión ambiental y facilitar la comunicación y el intercambio de información entre organismos ambientales.

VII.- El Acceso a la Información Pública y la actividad nuclear

Si bien en el ámbito de la actividad nuclear no existe legislación que establezca específicamente el derecho de acceso a la información pública, cabe destacar que las leyes citadas previamente resultan de aplicación en la materia.

En el ámbito nuclear, la confianza pública en el uso tecnológicamente seguro de los materiales y las técnicas nucleares está íntimamente relacionada con el historial del

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[19] http://www.hacienda.jujuy.gov.ar/legislacion_prov/leyes/4444.html
[20] http://www.jgm.gov.ar/archivos/AccesoInfoPub/Normativa/normativa_local/ProvinciaDeMisiones_InfAmbiental.pdf
[21] Ley N° 3.764. BO N° 6.629 del 6/11/1992
[22] Ley N° 8.803. BO de fecha 15/11/1999
[23] Ley N° 653. BO de fecha 3/01/2005.
[24] Ley N° 5.533. BO de fecha 15/9/2003.
[25] "El Acceso a la Información Pública Ambiental". Terzi S., Iribarren F.

organismo regulador en cuanto a la divulgación rápida, precisa y completa de información relativa a esas cuestiones y actividades[26] .

Por otro lado, cabe señalar que lo dispuesto en el Decreto PEN Nº 1172/03, resulta de aplicación a la Autoridad Regulatoria Nuclear, organismo autárquico en jurisdicción de la Presidencia de la Nación, dedicado a la regulación y fiscalización de la actividad nuclear que se realiza en toda la República Argentina, cuya finalidad, entre otras, es proteger a las personas del efecto nocivo de las radiaciones ionizantes[27].

En concordancia con lo establecido en el citado decreto, la Autoridad Regulatoria Nuclear, mediante Resolución del Directorio N° 67/2004[28], garantiza el derecho de acceso a la información pública, a través de la instrumentación de los procedimientos de acceso a la información y de elaboración participativa de normas y presentación de propuestas y opiniones.

VIII.- El Derecho a la Información y Participación Ciudadana en el Ámbito Nuclear

Como expresáramos en el punto precedente, las normas dictadas a nivel nacional alcanzan al derecho de acceso a la información relativo a la actividad nuclear.

La actividad nuclear, como la mayoría de las actividades realizadas por el hombre, produce efectos sobre el ambiente siendo por lo tanto alcanzada por los principios comprendidos dentro de las declaraciones ambientales internacionales mencionadas en el punto III del presente trabajo y en instrumentos jurídicos internacionales específicos de la materia.

De esta forma, el derecho de acceso a la información entendido como una herramienta necesaria para disipar los temores y prejuicios que la ciudadanía posee frente a determinadas actividades y como forma de generar confianza en las mismas, fue extendiéndose desde el derecho ambiental a otras ramas del derecho, entre ellas el derecho nuclear.

En este sentido, la normativa específica aplicable a la actividad nuclear se encuentra reflejada en la Convención de Seguridad Nuclear que, con relación a la construcción de nuevas instalaciones nucleares, en su artículo 17 establece, que se deberá *"…Consultar a las Partes Contratantes que se hallen en las cercanías de una instalación nuclear proyectada siempre que sea probable que resulten afectadas por dicha instalación y, previa petición, proporcionar la información necesaria a esas Partes Contratantes, a fin de que puedan evaluar y formarse su propio juicio sobre las probables consecuencias de la instalación nuclear para la seguridad en su propio territorio."*[29]

Por otra parte, la Convención Conjunta Sobre Seguridad en la Gestión del Combustible Gastado y Sobre Seguridad en la Gestión de Desechos Radiactivos, dispone respecto de los proyectos de instalaciones de gestión de combustible gastado, que las partes contratantes adoptarán las medidas necesarias con el fin de facilitar al público información sobre la seguridad de dicha instalación y consultar a las Partes

[26] "Manual de derecho nuclear". Carlton Stoiber, Alec Baer, Norbert Pelzer y Wolfram Tomhauser.
[27] Ley Nacional de la Actividad Nuclear Nº 24.804. Publicada en el B.O. N° 28.634 de fecha 25/4/97.
[28] Resolución del Directorio de la ARN N° 67/04 de fecha 7/9/2004.
[29] Aprobada por Ley N° 24.776. Publicada en el B.O. N° 28.624 de fecha 11/4/1997.

Contratantes que se hallen en las cercanías de dicha instalación, en la medida que puedan resultar afectadas por la misma, y facilitarles, previa petición, los datos generales relativos a la instalación que les permitan evaluar las probables consecuencias de la instalación para la seguridad[30].

Podemos citar, como ejemplo de participación ciudadana en el ámbito de la actividad nuclear, a la consulta pública realizada en el año 2006 por el Gobierno de Gran Bretaña, de conformidad con lo establecido en la Convención de Aarhus, respecto del rol de la energía nuclear en la generación de energía sin emisión de dióxido de carbono.

Asimismo, durante el año 2010, la Unión Europea realizó consultas para que los Estados parte, organizaciones privadas, industrias, ciudadanos y organizaciones ambientales expresen su opinión respecto de proyectos normativos sobre combustible gastado y residuos radiactivos.

Otro ejemplo de consulta pública, efectuada en el marco de la Convención de Aarhus, fue realizada el presente año por la empresa EDF Energy de Gran Bretaña, respecto de la construcción de una nueva Central Nuclear en Hinkley Point.

En Perú se realizaron también durante el año 2011, una serie de audiencias públicas a fin de que todas aquellas personas, entidades, empresas y quienes tengan algún interés particular en los proyectos de leyes de seguridad nuclear y de desechos radiactivos, tuvieran la oportunidad de expresar su opinión así como de proporcionar propuestas, comentarios u observaciones que permitan enriquecerlos.

Con respecto a nuestro país, es posible citar el caso jurisprudencial "Schroeder c/ INVAP S.E. s/amparo"[31] en el que la Corte Suprema de Justicia de la Nación, el día 6 de mayo de 2009, llamó a una audiencia pública de carácter informativo, en la cual las representaciones de cada una de las partes, fueron interrogados sobre diversos aspectos técnicos[32].

IX.- Conclusión

Las actividades antrópicas generan en el ambiente impactos de diferentes magnitudes, motivando la preocupación de los ciudadanos en su protección y cuidado. Con el nacimiento del concepto de desarrollo sustentable fue, en la normativa ambiental, que se consagraron los derechos de participación ciudadana y acceso a la información, para luego extenderse a otras ramas del derecho, entre ellas el derecho nuclear.

La participación ciudadana posibilita que las personas se informen y opinen responsablemente acerca de un proyecto, política o norma específica. Es en los casos en que la ciudadanía presenta dudas y desconocimiento respecto de los niveles de seguridad o de la relación costos - beneficios de una actividad, en que la información clara y transparente brindada por los organismos gubernamentales y la participación ciudadana son aún más importantes.

[30] Aprobada por Ley N° 25.279. Publicada en el B. O. N° 29.455 de fecha 4/8/2000.

[31] S.C.S. 569, L. XLII

[32] La controversia giraba en torno a si la cláusula 3.2.3.2 del contrato suscripto entre la empresa INVAP S.E. y la Australian Nuclaer and Technology Organization era violatoria del artículo 41 de la Constitución Nacional.

Así, la inclusión de los derechos de acceso a la información y de participación ciudadana, en las convenciones nucleares constituyen un gran paso en pos del desarrollo sustentable de la actividad y del acercamiento de la ciudadanía a la misma, dado que una mayor participación ciudadana, la transparencia y el contacto fluido entre los habitantes y las autoridades regulatorias, contribuyen a una mejor aceptación, de las decisiones finales.

Es por esto, que tanto la participación ciudadana como el acceso a la información son herramientas útiles para que los ciudadanos se interioricen respecto de las actividades nucleares que se desarrollan en el país, así como para vencer los temores y prejuicios que poseen respecto de la misma.

En conclusión, si bien el marco jurídico nacional de la actividad nuclear no contempla específicamente los derechos de acceso a la información pública y de participación ciudadana, el sistema legal creado a partir de la reforma constitucional de 1994, la influencia del derecho internacional y los principios consagrados en las convenciones nucleares, constituyen el marco legal aplicable de los derechos mencionados en la actividad nuclear en nuestro país.

Bibliografía

- “Public Participation Regarding the Elaboration and Approval of Projects in the EU after the Aarhus Convention” T.F.M. Etty, H. Somsen (eds.), Yearbook of European Environmental Law, Vol. 4, Oxford University Press, Oxford 2004.
- “Environmental Protection under Nuclear Law: Still a Long Way to Go” S. Emmerechts. OECD Publications. France 2010.
- “Régimen de libre Acceso a la Información Pública Ambiental” L. Belfer, L. Editorial Astrea.
- “El derecho de Acceso a la Información Pública Ambiental en el Sistema Jurídico Nacional. Sanción de la Ley de Presupuestos Mínimos N° 25.831.” Sahaniuk, A. El Dial-Suplemento Ambiental.
- “Derecho de Libre Acceso a la Información Pública”, Nápoli, A.M. y Vezzulla, J.M.
- “Participación y Medio Ambiente”, Diez Arregi, P.
- “El Acceso a la Información Pública Ambiental”. Terzi, S.; Iribarren F.
- “Manual de derecho nuclear”. Carlton Stoiber, Alec Baer, Norbert Pelzer y Wolfram Tomhauser.

LEGAL FRAMEWORK RELATED TO ACCESS TO INFORMATION AND PUBLIC PARTICIPATION ON NUCLEAR ACTIVITY

– Abstract

The right of access to information by citizens about activities related to scientific and technological development of nuclear energy for peaceful uses, has evolved over time. Governments began to perceive the necessity and the benefits of informing the community, who manifested certain prejudices about nuclear activity as a consequence of the propelling of nuclear bombs in Nagasaki and Hiroshima.

With the advent of environmental law and the influence of its principles, the idea of transparency of information in the nuclear field was imposed, and also the importance of both the inhabitants of countries with nuclear developments and neighbouring countries who may be affected by the bordering effects of ionizing radiation, could have access to information and to participate actively.

The access to information and citizen participation has been institutionalized and reflected in international regulations through international conventions subscribed by our country and nationally through the National Constitution, the Provincials Constitutions, the City of Buenos Aires Constitution, Laws No. 25.675, 25.831 and PEN Decree No. 1172/03, among others

The present work aims to make an overview of the legal framework related to access to information on nuclear activity.

Infrastructure Upgrade Experience of Station IS02, Ushuaia, Argentina

Avaro, E.; Fernández, M.; Figueroa, C.;
Pantin, A.; Quintana, E. and Vigile, S.

Presentado en: Infrasound Technology Workshop 2011.
Jordania, 30 de octubre al 4 de noviembre de 2011

Nuclear Regulatory Authority
Argentina
www.arn.gob.ar

Infrastructure Upgrade Experience of Station IS02, Ushuaia, Argentina

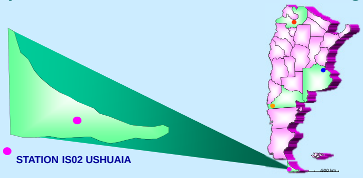
Avaro Ezequiel ; Fernández Marcelo; Figueroa Carlos; Pantin Andrés; Quintana Eduardo; Vigile Sebastián

Nuclear Regulatory Authority (ARN) - Argentina

INTRODUCTION

IS02 infrasound station is located 100 km from Ushuaia city, Argentina. It became operational in April 2005, but due to flooding of the vaults all of them were reinforced in January 2006. The station was certified in August 2006. After two years new problems with water filtration appeared at the station, making it extremely necessary to update its infrastructure.

The map shows facilities of the CTBT IMS in Argentina



WORKS IN CHRONOLOGICAL ORDER

Water Inside Underground Vaults and Pipes, November 2008


Equipment vault with water.


Manifold with water.


Battery vault with ice.

Corrective Maintenance Performed, January 2009


Manual removal of water from battery vault.


Pumping out water from pipes through the manifold.


Approx. 15 liters of water were extracted from pipes.

Step 1: Equipment Dismounting, January 2010


Batteries dismounted from the vault.


Transportation of dismounted equipment.


Equipment stored at the Central Facility.

Step 2: Demolition of All Sites, Early February 2010


Demolition of old equipment vaults.


Extracting GPS antenna pipe.


Leveling of ground sites with heavy machines.

Step 3: Civil Works In All Sites, Late February 2010


Ground preparation for concrete foundation.


Steel reinforced concrete frame for foundation.


Finishing of concrete foundation.


Installation of new GPS antenna pipe.


Gravel to prepare the bed for new pipes.


Prepared site to install new configuration pipes.

Step 4: Installation of Pipes and Vaults, March 2010


Pipeline and goosenecks layout.

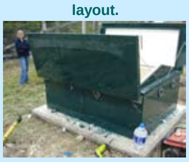

Connection of new rosette geometry.

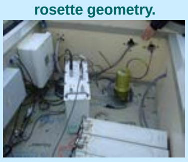

Checking leaks in pipeline connections.

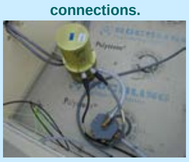

New pipe array geometry uncovered.

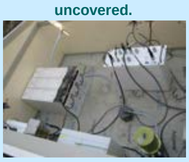

Inlets installed and covered with gravel.


New pipe array finished.


Installation of the new PVC equipment vaults.


Vault equipment layout.

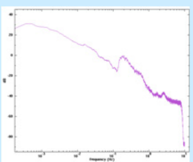

MB2000 microbarometer with sound suppressor.


Equipment installed inside the vault.

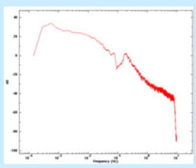

New PVC vault open with all the connections done.


Vault closed and sending data to PTS.

CONCLUSIONS


Before upgrade (with water)

From the station operator's point of view, the new design of the station sites proved to be successful since no water problems appeared in the vaults and pipes of all of them.
The graphs show the improvement in the signal in one of the arrays before and after the 2010 infrastructure upgrade.
The efforts made by PTS and Station Operator (ARN) allowed the achievement of the planned goal, resulting in a better behavior of the station since the works described in this poster were carried out.


After upgrade (without water)

For more information submit an e-mail to ctbtlab@arn.gob.ar

Operator Experience of HPGE Cryo-Cycle Detector Installation at the ARP03 Station

Avaro, E.; Fernández, M.; Pantin, A. and Quintana, E.

Presentado en: San Diego Operation & Maintenance Workshop.
San Diego, EE. UU., 9 al 13 de mayo de 2011

Nuclear Regulatory Authority
Argentina

www.arn.gob.ar

OPERATOR EXPERIENCE OF HPGE CRYO-CYCLE DETECTOR INSTALLATION AT THE ARP03 STATION

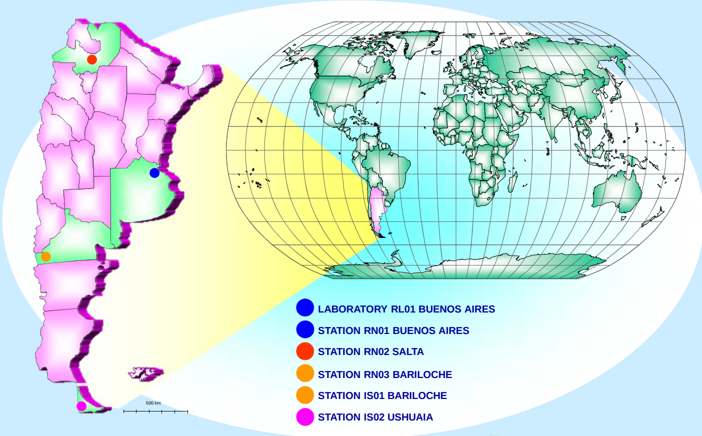
Avaro Ezequiel - Fernández Marcelo - Pantin Andrés – Quintana Eduardo

Nuclear Regulatory Authority (ARN) - Argentina

INTRODUCTION

ARP03 radionuclide station is located in San Carlos de Bariloche city, Argentina. It became operational in January 2004 and certified in July of the same year. The detector installed at the station was a Cryo-Electric II Cooling Compressor. In May 2008, a Canberra Cryo-Cycle Cryostat Germanium detector replaced the original installed.

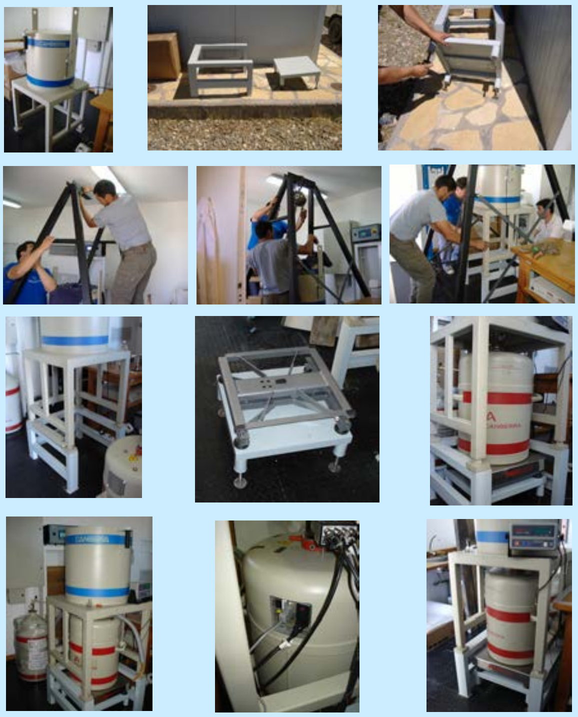
The map shows facilities of the CTBT IMS in Argentina



The main reason for the detector substitution was the reliability of the cooling system.

The Cryo-Electric II Cooling Compressor presented occasionally problems from the beginning and at the end very frequently.

Process of mounting and modifications made at the station to install a new detector system.



Cryo-Electric II Cooling Compressor

Cryo-Cycle Cryostat Germanium Detector

❖OTHER ALTERNATIVES CONTEMPLATED

- A new Cryolectric II Cooling Compressor
  Not good experience.
- Cryo-Cycle™ Hybrid Cryostat
  Experimental.
- Cryo-Pulse™ 5 Electrically Cooled Cryostat
  Experimental.
- Standard LN2 refrigeration
  LN2 logistic supplier problem.

Among the alternatives analyzed the chosen system became the solution to the problems.

❖ADVANTAGES OF THE HPGE CRYO-CYCLE DETECTOR:

- No lose of LN2 with cryocooler on
- Fiber-Carbon window
- 1 week cold conservation with cryocooler off

❖SUGGESTED IMPROVEMENTS FROM OPERATOR EXPERIENCE:

- More accurate LN2 level indicator
- USB interface for monitoring from a PC the internal pressure
- Internal preamplifier to get a vertical position / to avoid transversal tension

CONCLUSION

From the station operator's point of view, the choice of this type of cooling system provided a more reliable detection system than the previous one. One of the biggest and more common problems that stations have in the network are detector cooling difficulties / failures. This alternative could be the solution for stations with similar problems.

For more information submit an e-mail to ctbtlab@arn.gob.ar

Avances sobre la temática de dispensa y exención en el campo regulatorio argentino

Muñiz, C.C. y Bossio, M.C.

Presentado en: XXXVIII Reunión Anual de la AATN.
Buenos Aires, Argentina, 14 al 18 de noviembre de 2011

AVANCES SOBRE LA TEMÁTICA DE DISPENSA Y EXENCIÓN EN EL CAMPO REGULATORIO ARGENTINO

Muñiz, C.C. y Bossio M.C.

Autoridad Regulatoria Nuclear
Argentina

ABSTRACT

With the aim of optimizing the regulatory effort in Argentina, the Nuclear Regulatory Authority (ARN) evaluated two worldwide concepts used in the radioactive waste management field: “Generic Exemption Levels” and “Generic Clearance Levels”. The objective of this paper is to present the progress made in the past two years in relation to these topics and to present the results of the specific requests received from users of radioactive material. Since the approval of both Generic Levels, the ARN received two exemption requests. The first one, regarding the practice of dismantling lighting rods with ^{241}Am. The other case regards the international trade, distribution, usage and final disposal of lighting products with radioactive material (85Kr and 232Th). Concerning clearance, there has not been any request yet. However, in the future the ARN expects to receive this kind of requests from nuclear power plants and other facilities related to the nuclear fuel cycle.

Key words: Generic Exemption Levels, Generic Clearance Levels.

OBJETIVO

El objetivo del presente trabajo es informar los avances regulatorios realizados en el ámbito de Exención y Dispensa, entre ellos, la elaboración de guías regulatorias para facilitar la implementación de los Niveles Genéricos y los resultados de los casos específicos analizados.

INTRODUCCIÓN

Se entiende como Exención a la acción regulatoria mediante la cual se exime a determinado uso de una sustancia radiactiva, de los requisitos prescriptos en la normativa regulatoria. Mientras que al concepto de Dispensa se lo define como la liberación de material con contenido radiactivo que fue utilizado en prácticas licenciadas, autorizadas o registradas por la Autoridad Regulatoria, de la aplicación de todo control ulterior por parte de dicha autoridad.

A partir del 2007 y con motivo de mejorar la eficiencia de la gestión de fuentes y materiales radiactivos de muy baja actividad y/o concentración de actividad, la Autoridad Regulatoria Nuclear (ARN) comenzó a analizar la temática de Exención y Dispensa.

Al principio, las actividades consistieron en la recopilación y análisis de la bibliografía internacional vigente, entre ella, documentos desarrollados por el National Radiological Protection Board (NRBP) y por el Organismo Internacional de Energía Atómica (OIEA). Luego se analizaron los criterios dosimétricos básicos, las exenciones y dispensas genéricas junto con los escenarios utilizados para la derivación de sus niveles, las vías críticas de exposición, los parámetros utilizados y sus respectivos grados de conservadurismo. Finalmente, se evaluó la posibilidad de aplicar dichos valores en Argentina.

Como resultado del análisis efectuado, se propusieron Niveles Genéricos de Exención –NGE- y Niveles Genéricos de Dispensa -NGD- que fueron elevados al Directorio de la ARN para su aprobación. En ambos casos se dio curso favorable a la incorporación de los valores al sistema regulatorio, permitiendo que se continuara con el desarrollo de estos conceptos, especialmente a través de guías regulatorias para la implementación de dichos niveles.

Actualmente los Niveles Genéricos de Exención y Dispensa, junto con sus correspondientes Guías Regulatorias, se encuentran aprobados por Resolución del Directorio de la ARN, hallándose estas últimas publicadas en la Web.

DESARROLLO

Niveles Genéricos de Exención

Los NGE fueron establecidos por el Organismo Internacional de Energía Atómica en la Norma Básica de Seguridad BSS 115, y están basados en los modelos dosimétricos del documento "Radiation Protection 65" de la Comisión de la Comunidad Europea.

La ARN comenzó el análisis de los NGE en el año 2007. Para ello, se estudiaron varios documentos internacionales relacionados con esta temática y en particular, el documento base mencionado anteriormente. El estudio consistió en el análisis del grado de conservadurismo de los parámetros utilizados, la factibilidad de aplicación y representatividad de los mismos. Los resultados de la evaluación fueron expuestos en el Congreso Anual de la AATN del 2007.

En Enero del 2008 se elevó al Directorio de la ARN el informe final de la evaluación para que se considerara la incorporación de estos niveles al sistema regulatorio. El informe constaba de un cuerpo principal con el análisis detallado de los NGE para fuentes radiactivas, un listado con los NGE para 300 radionucleidos de uso común y un apéndice conteniendo los modelos dosimétricos y los cálculos de los niveles.

Luego de que el informe fuera aprobado, se comenzó con el desarrollo de una Guía Regulatoria para facilitar la implementación de dichos niveles. En Julio de 2010, el Directorio la aprobó por Resolución ARN N°48/10, bajo el nombre "Guía AR 6-R0, Niveles Genéricos de Exención" y ya se encuentra publicada en la Web.

Dicha Guía consta de 5 secciones y una tabla, a saber:

- Prefacio
- Explicación de términos
- Consideraciones generales
- Vías críticas de exposición para NGE en concentración de actividad
- Vías críticas de exposición para NGE en actividad
- Tabla con los valores redondeados de los NGE (en actividad y en concentración de actividad) junto con la correspondiente vía crítica de exposición.

Dentro del prefacio se establece que "*la ARN puede eximir a determinados usos de sustancias radiactivas de los requisitos prescriptos en la normativa regulatoria, en la medida que satisfaga el criterio de exención*" (punto 2 de la norma AR 10.1.12 Norma Básica de Seguridad Radiológica"). Además se establece que la guía se aplica a usos justificados de sustancias radiactivas y que no tiene carácter obligatorio. En la segunda sección se explican los términos: exención y cantidad moderada de material. Dentro de las consideraciones generales se detalla a qué usos se aplica la guía y cuándo no es aplicable, qué escenarios han sido considerados para la derivación de los niveles, las consideraciones para aquellos radionucleidos que forman cadenas de decaimiento y las condiciones que deben cumplirse en el caso que el uso de sustancias radiactivas emplee más de un radionucleido. Dentro de las dos últimas secciones se describen las vías críticas de exposición.

Desde que la ARN adoptó los NGE, se recibieron dos consultas sobre casos específicos de exención. El primero, fue un caso de desmontaje de pararrayos conteniendo ^{241}Am donde se solicitó la evaluación de los riesgos radiológicos asociados a la disposición de estos dispositivos como residuo común. Se concluyó que no correspondía otorgar la exención debido a que el material superaba los NGE y los criterios dosimétricos (10µSv/a).

El otro caso fue una consulta sobre aplicación de exención en la importación, exportación, transferencias intermedias hasta su uso por parte del público y disposición final de lámparas de luminaria con material radiactivo (^{85}Kr y ^{232}Th). Como se explica en la Guía Regulatoria AR 6, la importación y exportación de material radiactivo no están dentro del alcance de la exención. Se está evaluando la aplicabilidad de la exención para las etapas intermedias de comercialización, el uso y disposición final de dichas lámparas.

En el futuro se espera profundizar el análisis sobre la posibilidad de aplicar la exención condicional en aquellos casos en que, aunque se excedan los NGE, la actividad y/o concentración de actividad sea suficientemente baja como para cumplir con el criterio dosimétrico (10µSv/a). Esto podría darse en diversos productos de consumo masivo, por ejemplo la mayoría de los modelos de detectores de humo que se comercializan.

Niveles Genéricos de Dispensa

Los Niveles Genéricos de Dispensa fueron establecidos en la Guía de Seguridad del Organismo Internacional de Energía Atómica RS-G-1.7 y derivados a partir de escenarios desarrollados en la Guía de Seguridad N°44 del OIEA.

Con el fin de responder a una necesidad planteada por operadores de instalaciones nucleares y en el marco de la mejora del sistema de gestión de residuos, en el 2009 la ARN analizó los NGD con el fin de incorporarlos al sistema regulatorio.

El análisis de los NGD consistió en la evaluación de la documentación internacional vigente, de los escenarios y las vías de exposición utilizados para la derivación de los mismos, así como en el grado de conservadurismo de los parámetros utilizados. A principios del 2009 los resultados de la evaluación se elevaron al Directorio de la ARN, aprobándose los Valores Genéricos de Dispensa por Resolución N° 154/09. En Noviembre de ese año los resultados se presentaron en el Congreso Anual de la AATN.

Al igual que en exención, se comenzó con el desarrollo de una Guía Regulatoria que contemplara los criterios generales para la implementación de los valores genéricos aprobados. Finalmente, el 16 de Marzo de 2011 se aprobó dicha guía bajo el nombre “GUIA AR8- R0, Niveles Genéricos de Dispensa”, ya disponible en la Web.

La Guía consta de 8 secciones y una tabla, a saber:

- Prefacio
- Explicación de términos
- Referencias
- Alcance
- Radionucleidos de origen natural
- Radionucleidos de origen artificial
- Mezcla de radionucleidos de origen natural y artificial
- Determinación de la concentración de actividad en el material.
- Tabla con los valores de dispensa en concentración de actividad

En la primera sección se hace una introducción a la guía y se aclara que no es de uso obligatorio. En la segunda sección se explica el término “dispensa”. En la cuarta sección se detallan aquellos casos en los que son o no aplicables las recomendaciones de la Guía. En la quinta y sexta sección se detallan las consideraciones a tener en cuenta para radionucleidos de origen natural y artificial respectivamente. En la séptima sección se establecen las condiciones que deben cumplirse en el caso de tratarse de mezclas de ambos tipos de radionucleidos. Finalmente, en la octava sección se detalla qué debería contemplar el procedimiento para promediar la determinación de la concentración de actividad en el material y lineamientos generales de cómo efectuarlo.

Cabe aclarar que la ARN podrá autorizar dispensas para materiales que contengan una concentración de actividad superior a la de los niveles genéricos, en base a un estudio caso por caso, si se demostrara que esa es la mejor opción.

Si bien hasta el momento no se han otorgado dispensas de materiales radiactivos provenientes de instalaciones nucleares, se están analizando algunos casos que involucran grandes cantidades de residuos que podrían estar en condiciones de ser dispensados, con el consiguiente beneficio que se obtendría respecto a la optimización de recursos.

En ese sentido, otro objetivo a futuro de la ARN es analizar la posibilidad de aplicar valores de dispensa para contaminación superficial y presentar una propuesta al Directorio para la incorporación de los mismos al sistema regulatorio de la ARN.

CONCLUSIONES

Los avances realizados en el campo de exención y dispensa han sido muy productivos para Argentina ya que cubrieron un vacío regulatorio en el tema, permitiendo la optimización de los recursos económicos y humanos destinados a la gestión de material radiactivo.

Así mismo, esta incorporación nos posiciona en concordancia con las nuevas revisiones de la BSS 115 que introduce como novedad los niveles de dispensa complementando a los niveles genéricos de exención ya aceptados en versiones anteriores. Se adopta entonces una postura más armonizada con la comunidad internacional.

Además, ambos niveles genéricos son de gran utilidad para los usuarios de fuentes radiactivas y operadores de instalaciones radiactivas. En el primer caso, servirán para permitir el libre uso de ciertos materiales radiactivos que por su bajo nivel de riesgo no se justifica controlar, y en el segundo caso, para dispensar materiales de operación, de extensión de vida de las centrales y de desmantelamiento, cuando se llegue a esa etapa.

En el futuro la ARN espera poder evaluar casos específicos de exención en donde se sobrepasen los NGE, como es el caso de los detectores de humo y otros productos de consumo masivo. Asimismo, se espera analizar casos de dispensa de materiales provenientes de diferentes instalaciones radiactivas, principalmente las relacionadas con el ciclo combustible dada la gran cantidad de materiales radiactivos que generan y a la par seguir avanzando en el desarrollo en el tema desde el punto de vista regulatorio, especialmente a través de la derivación e incorporación de valores de contaminación superficial para la dispensa.

BIBLIOGRAFÍA

1- INTERNATIONAL ATOMIC ENERGY AGENCY. "Application of the Concepts of Exclusion, Exemption and Clearance". Safety Guide N° RS-G-1.7. IAEA Safety Series. Vienna, IAEA, 2004.

2- INTERNATIONAL ATOMIC ENERGY AGENCY. "Derivation of Activity Concentration Values for Exclusion, Exemption and Clearance". Safety Reports Series N° 44. Vienna, IAEA, 2005.

3- UNITED NATIONS. "Sources and Effects of Ionizing Radiation". Scientific Committee on the Effects of Atomic Radiation. New York, UN, 2000.

4- ORGANISMO INTERNACIONAL DE ENERGÍA ATÓMICA. "Normas básicas internacionales de seguridad para la protección contra la radiación ionizante y para la seguridad de las fuentes de radiación".Colección de Seguridad N° 115. OIEA, Viena, 1997.

5- INTERNATIONAL ATOMIC ENERGY AGENCY. "Application of Exemption principles to the recycle and reuse of materials from nuclear facilities". Safety Series N° 111-P.1.1. Vienna, IAEA, 1992.

6- COMMISSION OF THE EUROPEAN COMMUNITIES. "Principles and Methods for Establishing Concentrations and Quantities (Exemption values) Below which Reporting is not Required in the European Directive". Luxemburgo, CEC, 1993. Radiation Protection 65.

7- ASOCIACIÓN ARGENTINA DE TECNOLOGÍA NUCLEAR. "Análisis de los niveles genéricos de exención para fuentes radiactivas". M.C. Bossio, C.C. Muñiz. Autoridad Regulatoria Nuclear. AATN, 2007.

8- ASOCIACIÓN ARGENTINA DE TECNOLOGÍA NUCLEAR. "Análisis de los niveles genéricos de dispensa". M.C. Bossio, C.C. Muñiz. Autoridad Regulatoria Nuclear. AATN, 2007.

Concrete Reflected Cylinders of Highly Enriched Solutions of Uranyl Nitrate” ICSBEP Benchmark: a Re-Evaluation by Means of MCNPX Using ENDF/ B-VI Cross Section Library

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CONCRETE REFLECTED CYLINDERS OF HIGHLY ENRICHED SOLUTIONS OF URANYL NITRATE" ICSBEP BENCHMARK: A RE-EVALUATION BY MEANS OF MCNPX USING ENDF/ B-VI CROSS SECTION LIBRARY

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ABSTRACT

This work presents a theoretical re-evaluation of a set of original experiments included in the 2009 issue of the International Handbook of Evaluated Criticality Safety Benchmark Experiments, as "Concrete Reflected Cylinders of Highly Enriched Solutions of Uranyl Nitrate" (identification number: HEU-SOL-THERM-002)[4].

The present evaluation has been made according to benchmark specifications [4], and added data taken out of the original published report [3], but applying a different approach, resulting in a more realistic calculation model. In addition, calculations have been made using the latest version of MCNPX Monte Carlo code, combined with an updated set of cross section data, the continuous-energy ENDF/B-VI library. This has resulted in a comprehensive model for the given experimental situation. Uncertainties analysis has been made based on the evaluation of experimental data presented in the HEU-SOL-THERM-002 report.

Resulting calculations with the present improved physical model have been able to reproduce the criticality of configurations within 0.5%, in good agreement with experimental data. Results obtained in the analysis of uncertainties are in general agreement with those at HEU-SOL-THERM-002 benchmark document. Qualitative results from analyses made in the present work can be extended to similar fissile systems: well moderated units of ^{235}U solutions, reflected with concrete from all directions. Results have confirmed that neutron absorbers, even as impurities, must be taken into account in calculations if at least approximate proportions were known.

Key words: International Criticality Safety Benchmark Evaluation Project (ICSBEP), Criticality Benchmark Experiments, High-Enriched Solution Systems, criticality calculations with MCNPX, ENDF/B-VI validation.

1. INTRODUCTION

A total of 76 critical experiments were performed in the mid-1970s at the Rocky Flats Plant, which was operated at that time by Rockwell International. In those experiments, critical heights at room temperature were determined for high-enriched uranium solution in various containers and under various conditions of neutron reflection [1,2,3]. Fourteen of those critical experiments were considered acceptable for use as benchmark experiments, and so were included in the 2009 issue of the International Handbook of Evaluated Criticality Safety Benchmark Experiments, as "Concrete Reflected Cylinders of Highly Enriched Solutions of Uranyl Nitrate" (identification number: HEU-SOL-THERM-002) [4]. In these experiments, each involving a single reflected tank containing highly-enriched uranyl nitrate solutions, the critical height was determined by linear interpolation between slightly supercritical and slightly subcritical states. The tanks were cylindrical in shape and placed at different locations in a concrete reflector. Critical configurations had height-to-diameter ratios less than 1.2, and uranium concentrations varied between 59.65 and 334.77 grams per litre.

In the HEU-SOL-THERM-002 report an evaluation of experimental data was performed. In the calculation model presented in this document, several items were justifiably neglected, for the sake of simplicity. The effects on k_{eff} of omitting these items were calculated using the TWODANT two-dimensional code, with 44-group ENDF/B-V cross sections. The calculated effects were then summed to get corrections to the benchmark-model k_{eff} values, and combined quadratically as additional uncertainty. Also, in the same document, calculated k_{eff} values were given for each of the cases considered, for three different combinations of codes and cross sections: KENO with 16 Group Hansen-Roach models, KENO with 27-Group Scale models and, MCNP with continuous Energy ENDF/B-V cross section library.

This work presents a theoretical re-evaluation of the original experiments, according to benchmark specifications and added data taken out of the original published report [3]. In contrast with the analysis made in the benchmark itself, the present evaluation has been made applying a different approach: in the base cases no items have been omitted or neglected, and the calculated effects on keff values have been taken as uncertainties. Also, calculations have been made using the latest version of MCNPX Monte Carlo code [5,6], combined with an updated set of cross section data: the continuous-energy ENDF/B-VI library.

2. BENCHMARK EXPERIMENTAL SETUP

Highly-enriched solutions of uranyl nitrate [$UO_2(NO_3)_2$], dissolved in nitric acid and diluted to the desired concentration with water, were used for all of the cases. The uranium was enriched to about 93 weight percent in ^{235}U, and three uranium concentration were used: 59.65, 144.38 and 334. 77 g U per litre (low, medium and high concentration). Each solution contained some impurities, whose elemental concentrations were given with measured uncertainties of about 50%, what reflects the difficulty of measuring such small contributions. Of these, cadmium and boron are expected to be the strongest neutron absorbers. The critical height, at room temperature, was determined in each case by linear interpolation between slightly supercritical and slightly subcritical states.
The tanks used in the experiments were top-opened right circular cylinders. Each had a ~30 cm-long coaxial “tailpipe” of the same material welded to the bottom of the tank. These tailpipes passed the solution to and from the cylinder during experiments. Aluminium alloy AL-6061 cylinders of two diameters were used, as well as one diameter for stainless steel SS-304 cylinders.

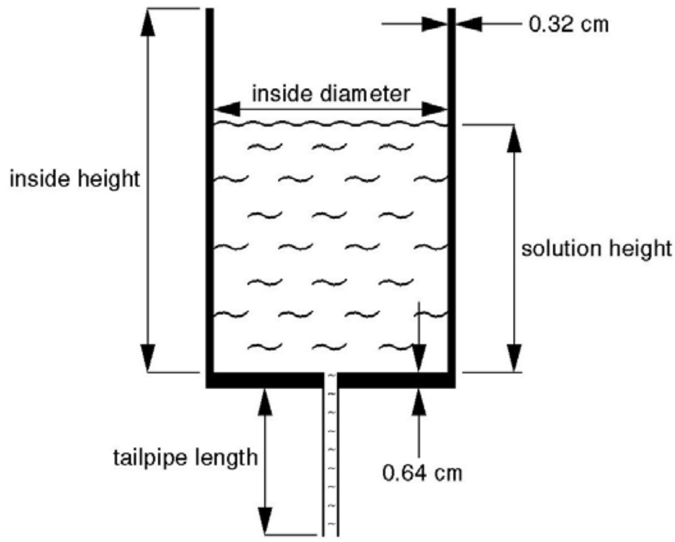

Figure 1: Experimental configuration.

The stainless-steel used had a density of 7.927 g/cm^3, and the aluminium alloy had a density of 2.737 g/cm^3. Because of nitric acid contents, a protective coating of acid-resistant paint (named "Phenoline 300") was applied to the inside of the aluminium tanks. Figure 1 is a sketchy representation of these tanks.

The tanks were located inside a concrete reflector, which geometrically was a thick-walled cubical shell of ~173 cm along its exterior side, with a ~122 cm interior cavity. The reflector was cast in six panels. Figure 2 shows a top view of the experimental configuration, while Figure 3 is a side view.

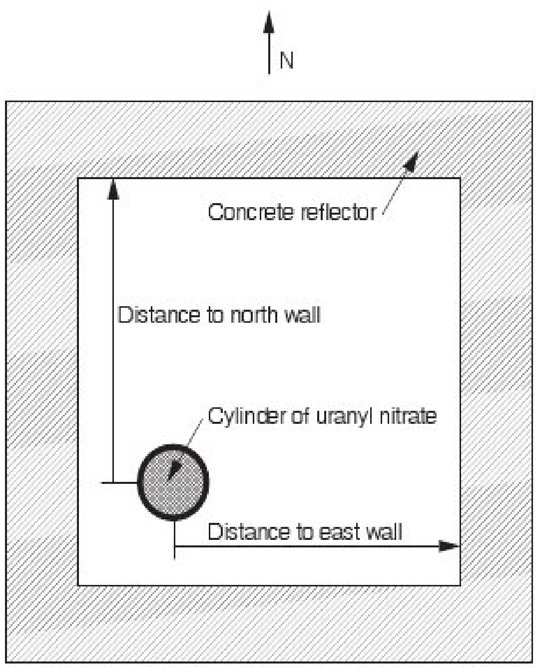

Figure 2: Top View of Experimental Configuration.

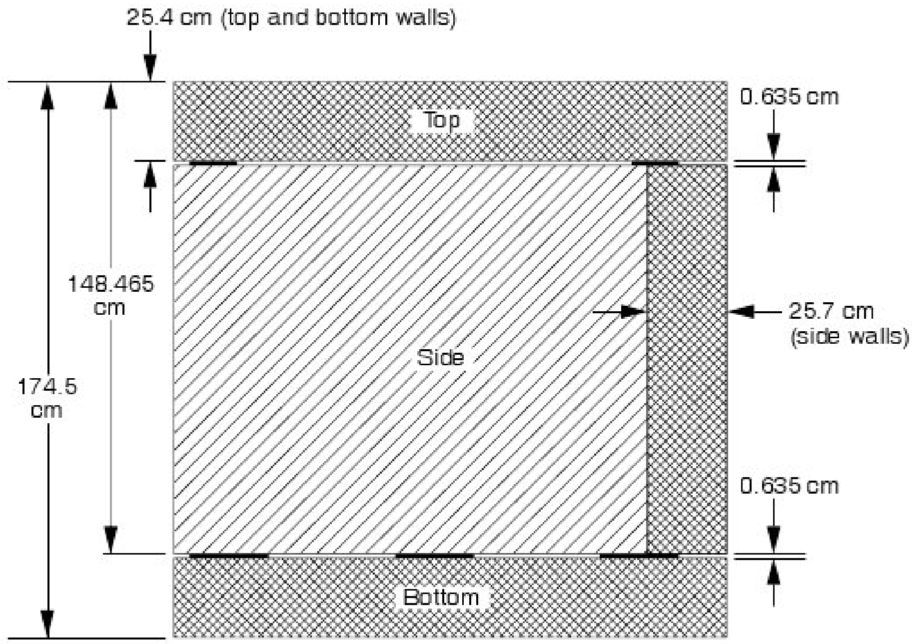

Figure 3: Side View of Concrete Reflector.

The top panel cracked during construction, so it was made safe by surrounding it with a steel compression band. Also in addition to concrete, the six panel combined contained ~11.9 kg of steel reinforcing wire and ~3.9 kg of other embedded steel pieces. This “re-bar” consisted of 0.48 cm-diameter steel rod welded in a rectangular grid-work placed in the mid-plane of each panel during pouring. Both top and bottom panels contained a given number of small holes serving various purposes (to receive the tailpipes, for example). The top had twenty-seven 2.5 cm-diameter holes, while the bottom had four of that diameter plus fourteen of half that size. In the same way each side panel contained one 3.8 cm-square hole at one corner, used as safety drain in the event of uranium solution leak. All holes and embedded material consumed less than 0.25% of reflector volume. All panels, except the top one, were lined with a 0.01 cm-thick vinyl sheet for contamination control.

A type of concrete representative of that used in the nuclear industry was selected for that set of experiments. The amount of water in the initial wet mix was about 9 weight %, being the elimination of this water the only change assumed in the reflector throughout the entire experiment. The experiments with the concrete reflector were performed approximately 4 to 8 months after the concrete was poured. Measurements showed that most of water weight loss occurred before the first experiments. Strong neutron absorbers boron, cadmium and chlorine were present in traces as impurities in concrete composition.

Figure 4 shows an elevation of the reflector concrete shell and its steel supporting structure. The stainless steel tank a few centimetres below the reflectors served as a distribution manifold directing solution to the cylinder as needed.

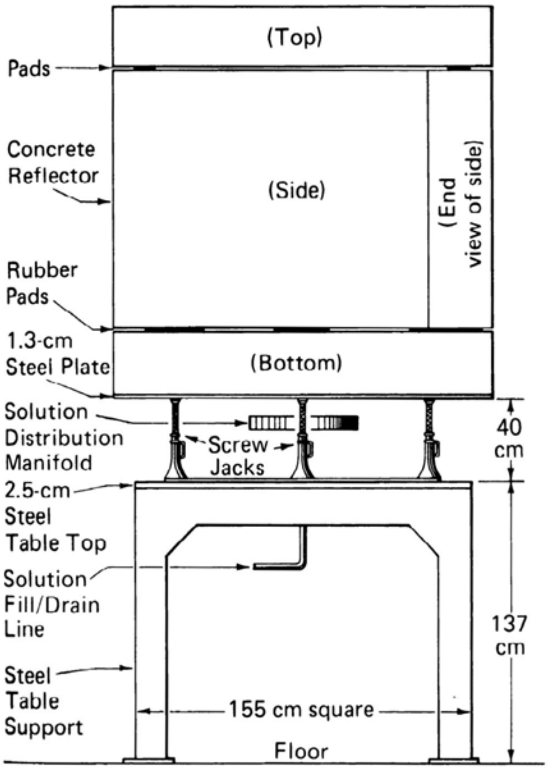

Figure 4: Schematics of the concrete reflector in elevation.

Table I summarise some important characteristics of the fourteen experiments carried out. More details may be found in reference documents [1,2,3,4].

Table I: Brief description of experimental cases.

Case number	Tank material	Inside diameter (cm)	Uranium concentration (g/cm^3)	Relative tank position
1	SS-304	27.92	144.38	Over the bottom panel
2	SS-304	27.92	144.38	On the bottom panel
3	SS-304	27.92	334.77	Over the bottom panel
4	SS-304	27.92	334.77	On the bottom panel
5	AL-6061	28.01	144.38	Over the bottom panel
6	AL-6061	28.01	144.38	On the bottom panel
7	AL-6061	28.01	334.77	Over the bottom panel
8	AL-6061	28.01	334.77	On the bottom panel
9	AL-6061	33.01	59.65	Over the bottom panel
10	AL-6061	33.01	59.65	On the bottom panel
11	AL-6061	33.01	144.38	Over the bottom panel
12	AL-6061	33.01	144.38	On the bottom panel
13	AL-6061	33.01	334.77	Over the bottom panel
14	AL-6061	33.01	334.77	On the bottom panel

3. MODEL USED AT HEU-SOL-THERM-002

In the HEU-SOL-THERM-002 report, an analysis of experimental data was made. In addition, two-dimensional calculations with TWODANT code, using 44-group ENDF/B-V cross sections from SCALE 4.4, were made for determining the effect of some items on reactivity. Using the results obtained, these items were neglected in order to simplify the calculation model.

In that model, the total measured impurities in solution were omitted, and the effect of including them was calculated for each case. Solution density and measured concentrations were conserved by reducing the water in solution by the mass of the impurity compounds.

Concerning tank materials, the measured weight percentages of all components were used for 304 stainless steel but, in the case of aluminium alloy 6061 only aluminium was used. The k_{eff} effect of adding the other measured constituents for each aluminium-tank case was calculated as an uncertainty, which was found to have a small positive effect and so omitted in the calculation model. Furthermore, the model was simplified by omitting the tailpipes containing fissile solution. As mentioned before, phenolic paint was used to protect the inner surfaces of the aluminium tanks from etching by the acidic solution. However, as remarked in references, tanks sometimes needed repainting due to the paint's gradually wearing away in the acidic solution. No phenolic paint layer was included in this calculation model, and its effect on k_{eff} was calculated by adding it to the initial model.

Regarding concrete, the re-bar region was assumed equivalent to a 0.49-cm-thick region with the given mass and volume of steel homogenized with the remaining volume fraction of concrete, and added at the mid plane of the walls in the models. The effect of adding this region was found to be negligible, so it was no considered in the model. In addition, the small holes at concrete reflector (for detector wiring, etc.) were simulated by reducing volume by approximately 0.25% and, as its effect was found to be negligible it was calculated as an additional uncertainty. The model also excluded the 0.01-cm-thick vinyl sheet of plastic lining the interior bottom and sides of the concrete reflector, which composition was given as that for polyvinylchloride, (C_2H_3Cl). It was assumed that its composition and thickness were not carefully measured, so that the effect of adding this plastic covering was calculated for all cases.

Besides, it was stated that a total of 2746 ppm of impurities were measured in concrete, being the only strong neutron absorbers: 24 ppm boron, 42 ppm chlorine, and 0.28 ppm cadmium, with no uncertainties given. Because of the difficulty of measuring such small amounts and because the main effect of impurities is expected to be from these three absorbers, the entire effect of adding the measured amounts of B, Cd, and Cl to the concrete was calculated as an standard uncertainty.

Finally, as the effect of steel plates below reflector was found negligible, the corresponding effect of adding them was considered as an additional uncertainty. The same result arose for solution distribution manifold and tailpipes welded to the bottom of the tanks.

4. MODIFIED CALCULATION MODEL

As described in the section before, in the calculation model developed at the HEU-SOL-THERM-002 report some items were neglected or omitted, for the sake of simplification. In contrast with this, the present model has been developed applying a different approach: in the base cases no items were omitted or neglected. This resulted in a more realistic calculation model for the given experimental situation. This model has been developed according to benchmark specifications [4], and added data taken out of the original published report [3].

In this model, the compositions for high-enriched uranyl nitrate solutions have been set up according to given data for uranium isotopic distribution, uranium concentration and excess nitric acid. All impurities have been taken into account and were assumed to be those given in references, apportioned according to their typical concentrations. Solution densities have been taken from the corresponding table in documentation.

With regard to tank dimensions, they have been assumed to be those given at tables and corresponding figures. For material composition, given at reference documents, all constituents have been taken into account for both 304 stainless-steel and aluminium alloy 6061. Besides, in all cases tailpipes of the same material have been added according to given geometry and dimensions, as well as phenolic paint of the corresponding thickness in aluminium tanks.

Concrete reflector has been modelled in accordance with given dimensions, and its composition has been assumed that measured after 16 months of curing, taking into account all neutron absorber given as impurities (boron, chlorine and cadmium). The re-bar present in all panels has been included as a 0.49-cm-thick region of homogenized concrete and steel, located at the mid plane of the panels. In addition, the 0.01-cm-thick polyvinylchloride (C_2H_3Cl) layer, lining the interior bottom and sides of the concrete reflector, have been considered as part of the model, being its density 0.91 g/cm^3, as stated in references. This more complete model has also included the steel plates below reflector, the compression band around top panel, and the solution distribution manifold.

Uncertainties analysis has been made based on the evaluation of experimental data presented in the HEU-SOL-THERM-002 report. A brief description of this analysis, together with the study of parameter effect is presented now.

According to references, the reported uncertainties related to solutions represent one standard deviation about the mean for multiple samples. Also, it is stated that the maximum relative uncertainties from the entire series of experiments for the high, medium, and low uranium concentrations were 0.6%, 0.7%, and 2.3%, respectively. In this work, effects of +1% changes in this concentration have been calculated and, following analysis made in HEU-SOL-THERM-002 results for standard uncertainties have been obtained by scaling the effect in order to match the given uncertainties (i.e., multiplying the calculated Δk_{eff} by 0.6/1 = 0.6, and so on). The effect of the uranium isotopic uncertainty has been calculated by increasing the ^{235}U weight percent by twice the standard deviation, and increasing the total of the weight percents of ^{234}U, ^{236}U, and ^{238}U by the same amount, proportioned according to their individual weight percent. The effect of the standard uranium isotopic uncertainty (reported to be one standard deviation) has been obtained by dividing the calculated results by 2.

The largest standard uncertainty in solution density (reported to be 0.0025 g/cm^3) has been taken as the standard uncertainty of the measurement method. The effect of a 0.002 g/cm^3 increase has been calculated for each case, and the k_{eff} effects of the standard uncertainty scaled to 0.0025 g/cm^3. In the same way, the largest measured relative uncertainty in nitric acid contents (reported to be 3.7%) has been taken as the estimate of the uncertainty of the measurement method for all cases. In order to estimate this effect on k_{eff}, a 2% increase in free nitric acid has been calculated for each case, with solution density conserved by decreasing the water. The results have been scaled to the standard uncertainty (i.e., multiplying the calculated Δk_{eff} by 3.7/2 = 1.85). The effect of removing all the nitric acid excess has been also calculated. In addition, the effect of omitting the total measured impurities has been calculated simply by excluding them from the model. As impurities served to both absorb neutrons and displace water, solution density and measured concentrations have been conserved by reducing the water in solution by the mass of the impurity compounds.

The effect of increasing the tank material density by 0.4 g/cm^3 on k_{eff} has been also calculated. Uncertainties in the reported, measured densities have been assumed to be 1 in the last reported digit, so the last results have been scaled to this value. Effect of tank thickness on k_{eff} have been calculated making changes in thickness of +0.1 cm for the side wall and +0.2 cm for the bottom, and scaled then to the standard uncertainties.

The effect of critical height on k_{eff} has been calculated by increasing it by 0.2 cm for each case. Taking into account the different uncertainties associated with critical-height measuring, HEU-SOL-THERM-002 report concluded that critical-height standard uncertainty was 0.11 cm for all cases, so results have been scaled to 0.11 cm in order to obtain k_{eff} standard uncertainty. Effect of solution radius on k_{eff} has been calculated increasing the solution radius by 0.2 cm, and then the results were scaled to the corresponding value for the solution-radius standard uncertainty for each case.

The effect of elimination of water on k_{eff} has been estimated by adding all initial water and, following reference [4], assuming an standard uncertainty equal to 1/5 the calculated effect of added water. Moreover, calculations with concrete density increased by 2% have been made to estimating the effect on k_{eff}, and the effect of a concrete-density decrease of 0.25% (reported as the standard uncertainty) has been obtained by scaling the results to this value.

The effect of moving the side walls closer to the tank by 1 cm has been calculated. The standard uncertainties for all cases have been calculated by scaling to the assumed uncertainty in position. Finally, calculations have been made in order to the estimate the effect of excluding the following items: vinyl lining from the interior of bottom and sides of the concrete reflector, measured impurities in concrete, compression band at top concrete panel, phenolic paint in aluminium tanks, solution tailpipes, solution distribution manifold, and steel plates below concrete. Also calculations with no concrete reflector were performed in order to determine the relative effects of this reflector.

In this work, all calculations have been made using the latest version of MCNPX Monte Carlo code (i.e. version 2.7E [5,6]). This code allows the development of a complete three-dimensional calculation model, using one of the most up-to-date comprehensive physical models for neutrons. In addition, the code has been combined with an updated set of cross section data: the continuous-energy ENDF/B-VI library. This combination, added to the mathematical model developed in this work, has resulted in a comprehensive representation of the real experimental situation. In MCNPX, effective multiplication factor (k_{eff}) is estimated in three different ways [7]: by collision, absorption, and track length estimators. In addition, these estimates are combined using observed statistical correlations to provide the optimum final estimate of k_{eff} and its standard deviation. This has been the estimate for k_{eff} used in this work.

5. RESULTS AND DISCUSSION.

Using the calculation method mentioned in the section before, the k_{eff}'s for the complete model (i.e. with no items omitted or neglected) have been calculated for each base case in table I. Effects from changes or omission of given items have been estimated by making the proposed variations, and calculating the corresponding k_{eff}'s.

Table II. Calculation results for k_{eff}

Case number	$k_{eff} \pm \sigma$
1	1.0002 ± 0.0066
2	1.0012 ± 0.0073
3	0.9951 ± 0.0081
4	0.9951 ± 0.0071
5	1.0001 ± 0.0075
6	1.0030 ± 0.0067
7	0.9979 ± 0.0045
8	1.0007 ± 0.0056
9	0.9954 ± 0.0095
10	0.9963 ± 0.0105
11	0.9989 ± 0.0069
12	1.0021 ± 0.0058
13	0.9962 ± 0.0053
14	1.0031 ± 0.0073

Experimental uncertainties of reported data and those arising when omitting certain items give rise to an uncertainty in the calculated k_{eff} values. As described in previous section, uncertainties from experimental data have been obtained by scaling the corresponding calculated effect. All estimated uncertainties have been combined with the calculated k_{eff}'s to obtain the benchmark-model one-σ uncertainties for k_{eff} 's. Final results are shown in table II, where resulting one-σ uncertainties include the quadratic combination with statistical uncertainties given by MCNPX.

Table III. Calculated effects on k_{eff}.

Parameter	Parameter variation calculated	Case 1	Case 2	Case 3	Case 4	Case 5	Case 6	Case 7
uranium concentration	+1%	-0.147	-0.057	-0.329	-0.354	0.077	0.057	-0.436
^{235}U weight per cent	+2 sigma	-0.108	0.082	0.131	-0.067	0.137	0.027	0.018
nitric acid	removed	0.808	0.903	1.305	1.122	1.072	0.914	1.276
nitric acid contents	+2%	-0.157	-0.009	-0.014	-0.133	0.153	0.004	-0.062
impurities in solution	removed	0.227	0.450	0.281	0.168	0.547	0.469	0.169
solution density	+0.002 g/cm3	0.016	0.002	0.250	0.216	0.199	0.161	0.078
critical height	+0.2 cm	0.081	0.350	0.356	0.323	0.247	0.252	0.236
solution radius	+0.2 cm	0.781	0.959	1.005	0.790	1.087	0.895	0.876
tank material density	+0.4 g/cm3	-0.042	0.050	0.133	0.064	0.147	0.129	0.032
tank wall thickness	+0.1 cm	0.300	0.373	0.490	0.336	0.271	0.124	0.208
tank bottom thickness	+0.2 cm	0.007	0.044	0.287	0.033	0.132	-0.042	-0.138
phenolic paint	removed	--	--	--	--	0.015	-0.095	-0.096
concrete density	+2%	-0.183	0.121	0.015	0.075	0.178	0.008	-0.025
water in concrete	initial added	-0.215	-0.030	0.027	0.028	0.082	0.008	-0.083
re-bar	removed	-0.149	0.052	0.032	-0.111	0.097	0.028	0.032
impurities in concrete	removed	-0.055	0.162	0.163	0.180	0.113	0.316	0.117
vinyl lining	removed	-0.142	0.069	0.072	0.111	0.087	0.001	0.078
compression band	removed	-0.096	0.063	0.146	-0.075	0.025	-0.011	0.015
concrete panels	removed	-1.522	-7.378	-1.598	-8.978	-1.436	-8.194	-1.911
tank position	1 cm closer	-0.133	0.088	0.035	-0.049	0.078	0.004	-0.048
steel plates below	removed	-0.116	0.031	-0.039	0.066	-0.013	-0.009	0.009
tailpipes	removed	-0.096	0.053	-0.039	0.007	0.119	-0.189	-0.019
solution manifold	removed	-0.176	0.084	-0.013	-0.087	0.047	-0.055	0.033

Table III. Continued

Parameter	Parameter variation calculated	Case 8	Case 9	Case 10	Case 11	Case 12	Case 13	Case 14
uranium concentration	+1%	-0.388	0.169	0.248	-0.045	-0.064	-0.463	-0.459
^{235}U weight per cent	+2 sigma	-0.003	-0.022	-0.016	0.121	-0.012	0.013	0.037
nitric acid	removed	1.162	0.493	0.549	1.042	0.911	1.187	1.148
nitric acid contents	+2%	-0.052	-0.087	-0.002	0.025	-0.057	-0.102	-0.077
impurities in solution	removed	0.112	0.818	0.756	0.432	0.309	0.116	0.174
solution density	+0.002 g/cm3	0.083	0.064	0.220	0.271	0.128	0.143	-0.012
critical height	+0.2 cm	0.215	0.076	0.267	0.474	0.392	0.311	0.476
solution radius	+0.2 cm	0.888	0.655	0.627	0.699	0.602	0.631	0.559
tank material density	+0.4 g/cm3	0.069	0.026	0.077	0.173	0.142	0.163	0.117
tank wall thickness	+0.1 cm	0.142	0.010	0.092	0.221	0.046	0.126	0.075
tank bottom thickness	+0.2 cm	-0.091	-0.001	-0.065	0.159	-0.136	0.161	-0.140
phenolic paint	removed	-0.104	-0.046	0.074	0.124	0.029	0.107	0.008
concrete density	+2%	0.025	-0.034	0.040	0.120	0.030	-0.083	0.195
water in concrete	initial added	0.198	-0.070	0.049	0.051	0.180	-0.009	0.366
re-bar	removed	-0.025	0.046	-0.008	0.075	-0.020	0.042	-0.061
impurities in concrete	removed	0.248	0.056	0.253	0.144	0.216	0.131	0.336
vinyl lining	removed	-0.118	-0.074	-0.074	0.064	-0.098	-0.075	-0.116
compression band	removed	0	0.005	-0.027	0.139	-0.033	-0.001	0
concrete panels	removed	-9.754	-1.344	-7.221	-1.636	-11.402	-1.843	-13.674
tank position	1 cm closer	-0.060	0.079	0.058	0	0.073	0	0.123
steel plates below	removed	-0.047	-0.013	0.011	-0.001	-0.021	-0.067	-0.022
tailpipes	removed	-0.242	-0.089	-0.108	-0.003	-0.218	-0.047	-0.354
solution manifold	removed	-0.114	-0.026	0.023	0.057	-0.033	0.047	-0.118

Results of effects on k_{eff} from changes or omission of given items, for the fourteen cases are summarized in Table III. The resulting effects have been calculated as the per cent difference with the associated k_{eff} for base cases.

The results obtained in this work will be discussed now. Because the effects related to solution properties are expected to be the more important, they will be analysed first. It has been found that meaningful variations in uranium isotopic composition produce no significant changes on k_{eff} (up to only 0.13%), so they may be neglected. On the other hand, it has been observed that meaningful variations in uranium concentration produce significant effects, with absolute values ranging from 0.05 to 0.46 %. In consequence, models should include a sensitivity analysis for this concentration. Concerning solution density, it has been found that significant variations increased k_{eff} up to about 0.27%, so it is an important effect to be included in models.

Removal of all nitric acid excess has been found to produce increasing of k_{eff} ranging from 0.5 to 1.3%. This effect on k_{eff} is meaningful, so it has been concluded that nitric acid excess must be included in the calculation model. In addition, it has come out that significant variations in this excess (+2%) produce small changes on k_{eff} (up to only 0.16%). Consequently, this variation may be neglected at a first approximation. In regard to impurities, it has been observed that its removal from solution produce a positive effect on k_{eff}, ranging from 0.11 to 0.82 %. References remarked that they represent small amounts, with measuring uncertainties of about 50% and main contribution coming from cadmium and boron. In spite of this, the effect is significant (is the main contribution to uncertainties in cases 9 and 10, as may be seen in table III), so it must be taken into account in the model, or in a sensitivity analysis.

Meaningful variations in critical height have been found to produce important changes on k_{eff} (up to 0.48%). In the same way, significant variations in solution radius have been found to have considerable effect on k_{eff} (ranging from 0.56 to 1.09%). Accordingly, effect of changes in both height and radius of solution should be studied.

It has been found that significant changes in both tank walls and bottom thicknesses produce meaningful effect on k_{eff} , up to 0.49% and 0.28% respectively. Tanks serve to both absorb and reflect neutrons, so the mentioned effects would depend on geometry and should be studied. On the contrary, it has been found that significant changes in tank material density produce no meaningful effects on k_{eff}, so density changes in both steel and aluminium may be neglected. The same result has been obtained for phenolic coating. Taking into account the coat thickness (about 180 μm), it is an expected result. In consequence the phenolic paint may be disregarded in the model.

Concerning impurities in concrete, its removal increased k_{eff} up to about 0.36%. As reflection is expected to be more important with the tank on the bottom panel, and impurities (mainly boron, cadmium and chloride) produce neutron absorption, the effect on k_{eff} is expected to be more significant in these cases; results have confirmed this fact. In regard to addition of initial water contents, it has been found that it produce meaningful effects on k_{eff} (up to 0.36%), more apparent in cases with tank on the bottom panel, as discussed below. Consequently, water content has to be taken into account in the model. Concerning density, it has been found that increasing it by 2% produce small, but meaningful changes on k_{eff} (up to 0.2%). Accordingly this effect must be taken into account if significant changes are possible on concrete density. The effect of moving tanks 1 cm closer to concrete panels has been found to be no more than 0.13% on k_{eff}, which represent only a small contribution to total uncertainties, so it may be neglected. Concerning vinyl lining on concrete panels, as expected because of its small thickness (0.01 cm) it has been found to have no significant effect on k_{eff}. The same result has arisen for re-bar, and for the compression band around the top panel. In consequence these three items may be neglected in the model.

By far the more important effect on k_{eff}, related to concrete, has been found when removing reflector panels. This removal decreased k_{eff} up to about 14%, being the effect more important when tanks were on the bottom panel, as expected because of discussion above. This result is in good agreement with reflecting properties of concrete on neutrons.

No significant effect has been observed when removing steel plates or solution distribution manifold below concrete reflector. Accordingly, they may be ignored in the model. Concerning tailpipes, its contribution was only significant (up to 0.35%) for aluminium tanks, when filled with medium and high enriched solution, and located on the bottom concrete panel.

6. CONCLUSIONS.

Benchmark-model has been re-evaluated using a mathematical model more realistic than that used in SOL-THERM-002 benchmark document. In addition, the most recent version of MCNPX Monte Carlo code has been used, combined with an updated cross section library, the ENDF/B-VI. All this has resulted in a comprehensive model for the given experimental situation. This model has been able to reproduce the criticality of configurations within 0.5%, in good agreement with experimental data.

The results obtained in the analysis of uncertainties are in general agreement with those at HEU-SOL-THERM-002 benchmark document. According to present results, the following items may be omitted, in order to simplify the calculation model: phenolic paint coating on aluminium tanks, vinyl lining on inside surface of concrete panels, steel compression bar around top panel, re-bar into concrete panels, and steel plates and solution distribution manifold below concrete reflector. These items are in agreement with those omitted in the model used at SOL-THERM-002 document.

Qualitative results from analyses made in the present work can be extended to similar fissile systems: well moderated units of ^{235}U solutions, reflected with concrete from all directions. Results has confirmed that neutron absorbers, even as impurities, must be taken into account in calculations if at least approximate composition were known.

7. REFERENCES.

1. R. E. Rothe and I. Oh, "Benchmark Critical Experiments on High-Enriched Uranyl Nitrate Solution Systems," *NUREG*/**CR-0041, RFP-2710**, Recoil International (April 1978).
2. R. E. Rothe and I. Oh, "Benchmark Critical Experiments on High-Enriched Uranyl Nitrate Solution Systems," *Nuc. Technol.*, **41**, 207 (1978).
3. I. Oh and R. E. Rothe, "A Calculational Study of Benchmark Critical Experiments on High-Enriched Uranyl Nitrate Solution Systems," *Nuc. Technol.*, **41**, 226 (1978).
4. International Handbook of Evaluated Criticality Safety Benchmark Experiments. NEA/NSC/DOC(95)03/II Volume II HEU-SOL-THERM-002: Concrete reflected cylinders of highly enriched solutions of uranyl nitrate.
5. MCNPX User's Manual. Version 2.6.0, April 2008 LA-CP-07-1473, Denise B. Pelowitz.
6. LA-UR-11-01502 MCNPX 2.7.E Extensions, March 2011.
   LA-UR-03-1987 MCNP — A General Monte Carlo N-Particle Transport Code, Version 5 Volume I: Overview and Theory, April 24, 2003.

Verification of Spent Fuel Assemblies (MTR) in Storage Facilities

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VERIFICATION OF SPENT FUEL ASSEMBLIES (MTR) IN STORAGE FACILITIES

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ABSTRACT

The Central Store for Special Irradiated Fissionable Material (DCMFEI) is an old Argentine facility, designed to store LEU spent fuel assemblies discharged from the research reactors. Spent MTR fuels are normally stored inside concrete wells filled with water and closed with lead plugs on the top. The water is used as shielding and cooling media. Each well can accommodate up to two fuel assemblies.

The facility is under the regulatory control of the Argentine Nuclear Regulatory Authority (ARN). On the other hand, the Brazilian-Argentine Agency for Accounting and Control of Nuclear Material (ABACC) and the International Atomic Energy Agency (IAEA) carry out safeguards inspections to this facility since the beginning of the implementation of the Quadripartite Agreement.

The spent fuel assemblies are maintained under dual containment system by ABACC and IAEA, which requires the verification of fuel assemblies at research reactors and to maintain the continuity of knowledge during the transfers to the storage. The transfer campaign takes one week and it is carried out once per year. In order to reduce the inspection effort is proposed to remove all seals. In this case, the Safeguards Criteria for Storage Facilities request item counting and verification of the spent fuel for gross defect.

Nevertheless, this task present some problems because the spent fuels are not always visible, then item counting is difficult. On the other hand, the introduction of detectors inside the wells is not always possible due to the small space available and the high doses involved when the wells remain opened.

In order to overcame these difficulties ARN and ABACC have designed and tested a method to verify spent fuel assemblies using gamma spectrometry. Spectra were taken with a Cadmium Telluride detector (CZT). An ad hoc modified lead plug was used as detector collimator. The activity of each item was determined trough the main Cs-137 peak. Experience showed that it is possible to distinguish among one, two or no spent fuel assembly. Moreover, the activity measured on each well shows good correlation with the theoretical activity calculated considering burnup, decay time and shielding corrections.

INTRODUCTION

The Central Store for Special Irradiated Fissionable Material (DCMFEI) is an old Argentine facility, designed to store spent fuel assemblies discharged from the research reactors. Spent MTR fuels are normally stored inside concrete wells filled with demineralized water and closed with lead plugs on the top. The water is used as shielding and cooling media. Each well can accommodate up to two fuel assemblies. The storage has 198 wells distributed in 6 rows of 16 wells each, and 6 of 17 wells each.

The wells in each row are interconnected and therefore all the wells in a particular row have the same water level. A pump recirculates the water in each row individually, so water from different rows never mix.

The facility is under the regulatory control of the Argentine Nuclear Regulatory Authority (ARN). On the other hand, the Brazilian-Argentine Agency for Accounting and Control of Nuclear Material (ABACC) and the International Atomic Energy Agency (IAEA) carry out safeguards inspections to this facility since the beginning of the implementation of the Quadripartite Agreement.

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Nevertheless, this task presents some problems because the spent fuels are not always visible, making item counting quite difficult. On the other hand, the introduction of detectors inside the wells is not always possible due to the small space available and the high doses involved when the wells remain opened. There is also some risk of damaging the fuel assemblies when detectors are introduced into the wells.

In order to overcome these difficulties ARN and ABACC have designed and tested a method to verify spent fuel assemblies using gamma spectrometry. This method should allow item counting and verification for gross defect, ensuring at the same time low dose levels for the inspectors.

The objective of this work is the development of a method for item counting and gross defect verification in this facility, taking into account the limitations previously explained. Consequently, for gross verification it is enough to verify qualitatively the existence of the Cs-137 peak (662 keV) in the well's spectrum and for item counting, it is necessary to distinguish among wells containing none, one or two fuel assemblies.

COLLIMATOR DESIGN

There had been several attempts to obtain the gamma spectra of the wells. The gamma spectra obtained with Sodium Iodide (NaI) and CZT detectors for closed wells show very few counts and do not give enough information to evaluate how many fuel assemblies are inside. In addition, when the measurement is carried out on a closed well it is not possible to ensure that the spectrum obtained is due only to the well. On the other hand, working with open wells involves high doses, especially when the well contains two fuel assemblies, because of the small thickness of water shielding. Even though a lead plate was used as a shielding over the opened well, the resulting dose values remained high and the obtained spectrum presented contributions from the neighbor wells. Besides these high doses, the introduction of a detector inside the well is not convenient due to the small space available and the risk of damage to the fuel assembly.

In order to overcome these difficulties ARN and ABACC designed a collimator. For this purpose a lead plug was modified as shown in Figure 1. As it was one of the plugs of the facility it fits perfectly into the top of the well. The lead plug has a 4mm Fe cover at its bottom. Two eyebolts were added to enable the crane to lift the plug. A hole of 25 mm in diameter was made in the center of the lead zone. This hole is large enough to contain the Cadmium Telluride detector inside. In the Fe plate the diameter of the hole is reduced to 20 mm. Thus, the detector placed inside the collimator can not fall. The detector has a total diameter of 24 mm, but the crystal only occupies an area of 10x10 mm in the center, so the plate of Fe does not contribute to attenuation. This design ensures that the spectrum obtained is due only to the well that is being measured, without introducing a detector in it.

Discs of 25 mm in diameter and different thickness were used as attenuators in different preliminary measurements. A lead collimator with a diameter of 5 mm was used in some measurements.

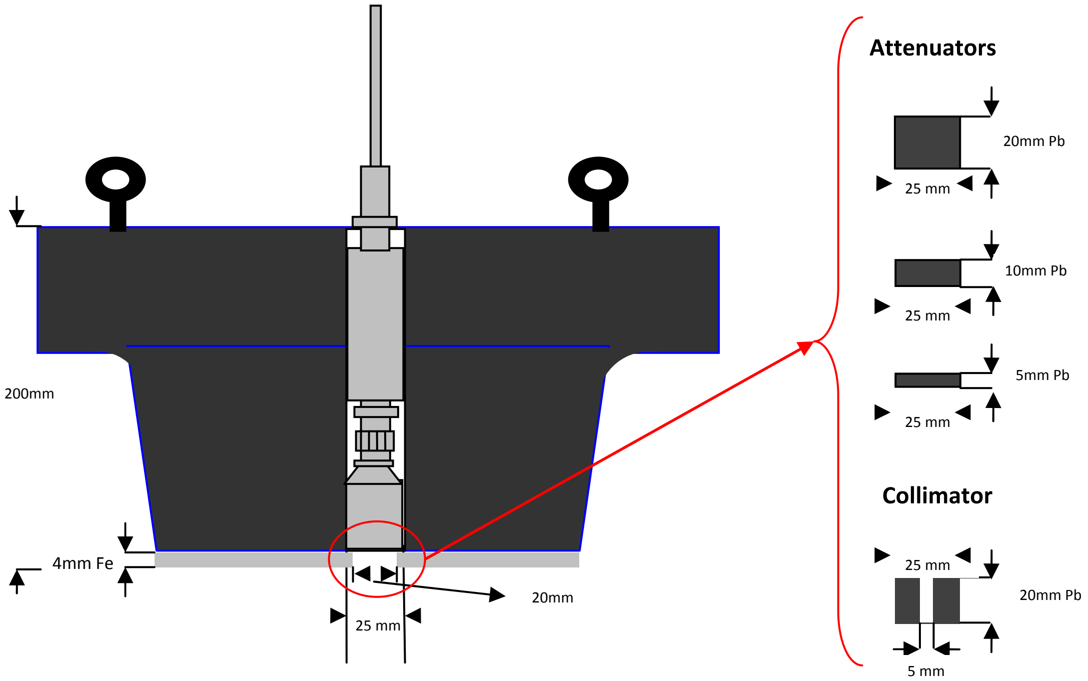

Figure 1: Collimator

FIELD MEASUREMENTS

One of the problems in the facility is the high doses involved when wells remain open. The method developed by ARN and ABACC for acquiring the spectra helps to avoid unnecessary exposure to high doses for inspectors.

The measurement points were selected considering two criteria. Firstly, a dosemetric evaluation of the facility was made using handheld radiation detectors in order to characterize the radiation field for selecting the measurements points: wells with high, medium and low background. And secondly, the U mass declared: wells with regular fuel assemblies (~1350 grams) and control fuel assemblies (~975 grams).

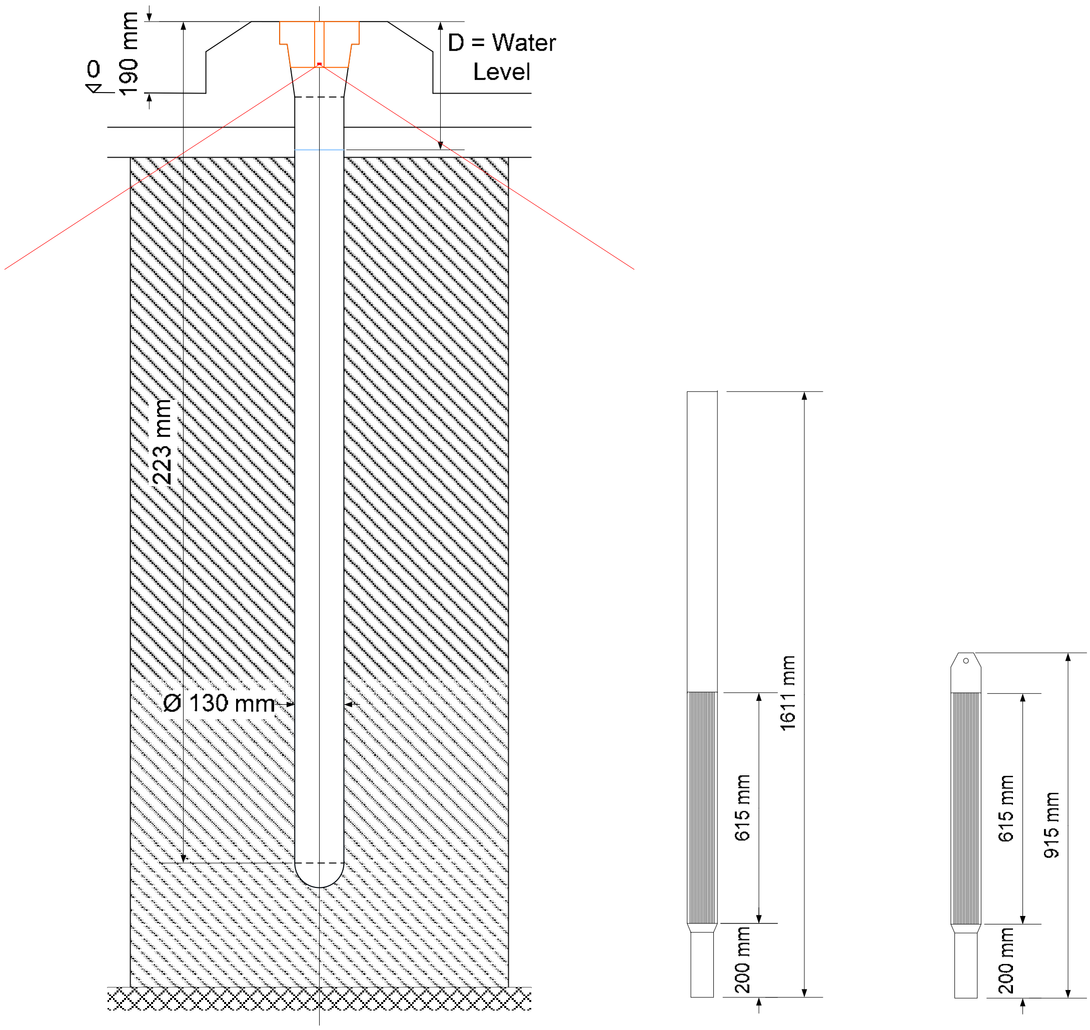

Figure 2: well profile, control and regular fuel assemblies schemes.

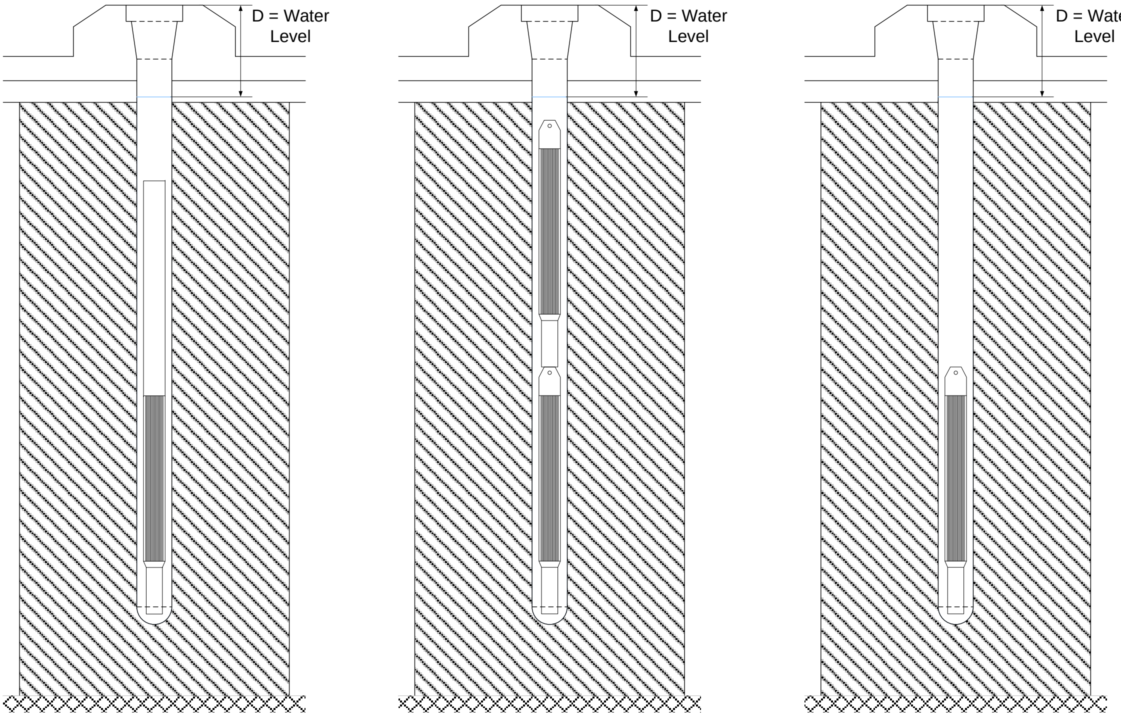

Figure 3: schematic for the nuclear material distribution inside the wells

The procedure to take the spectra is the following. Before opening the well, the detector was placed inside the collimator. Then, the crane operator removed the plug from the well and replaced it with the collimator containing the CZT detector. During this operation, all staff was required to move behind a lead wall by the radiation protection procedure. Once the collimator plug was properly seated, the personnel were allowed to leave the positions behind the lead shielding.

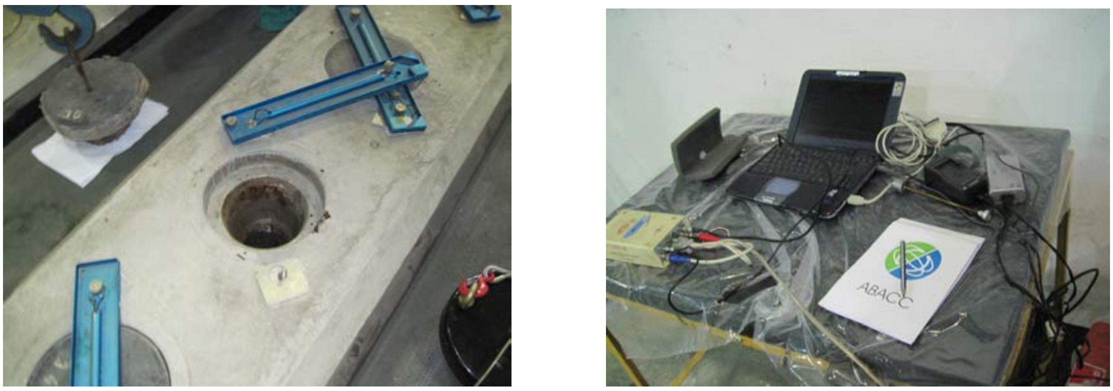

Figure 4: (a) Open well. (b) Equipment used for Gamma Spectra Acquisition.

The CZT detector was connected to the mini multichannel analyzer (MMCA) GBS MCA-166 by a 20 meters long cable. The MMCA and the notebook were located in a safe place, away from the well. The detector used was a Ritec Cadmium Zink Telluride CZT500(S) with a crystal of 10×10×5 mm and a guaranteed resolution (FWHM) of less than 18KeV at 662 keV. The gamma spectrum was acquired using the program WinSpec v1.03.0001, with 512 channels configuration. A region of interest of 33 channels (85.2 keV) around the 662.7 keV peak of Cs-137 was considered. After the acquisition, the collimator was removed and replaced by the original plug. At this moment the staff accomplished with the radiation protection procedures. During the replacement of the plugs, the dose was measured at the personnel location, behind the lead wall. No significant variation from dose background was observed.

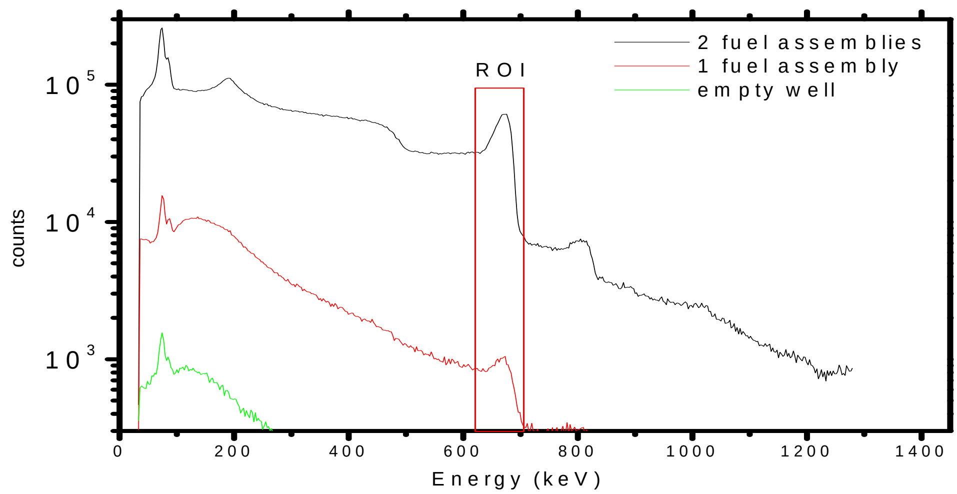

Figure 5: Spectra from a well containing 2 fuel assemblies, a well containing one fuel assembly, and an empty well.

One of the problems observed was the contribution to the background of water contaminated with fission products, particularly Cs-137 in some rows. There is also some remaining contamination on the walls of the well. This contamination takes place when a damaged fuel assembly is stored in a well. All the wells in a particular row are interconnected and the water is periodically recirculated and passed through ion exchange resins. The water contamination in a particular row is uniform, so the background is the same for all well in that row. The water of the different rows does not mix. Taking into account that the level of contamination depends on the fission products concentration and the level of water, the background of each row differs from each others. The last well of each row has no fuel assemblies and is considered as a “control well”. The control well is used to test the quality of water and to determine the background spectrum of the row. The gamma spectrum of the control well was acquired and applied as background for the corresponding row. The water level in the control well was also measured for shielding correction calculations.

DATA EVALUATION AND METHOD PROPOSAL

The method proposed is the following. Acquire the spectrum from the selected well and determine the numbers of counts in the region of interest around the Cs-137 peak (W). In order to obtain a background acquire the control well spectrum and determine the counts number in the ROI (B). Then, calculate A=W-B. Finally, compare A with the Table 1 to determine if the well contains one, two or none fuel assemblies. Figure 6 shows a flowchart of the method.

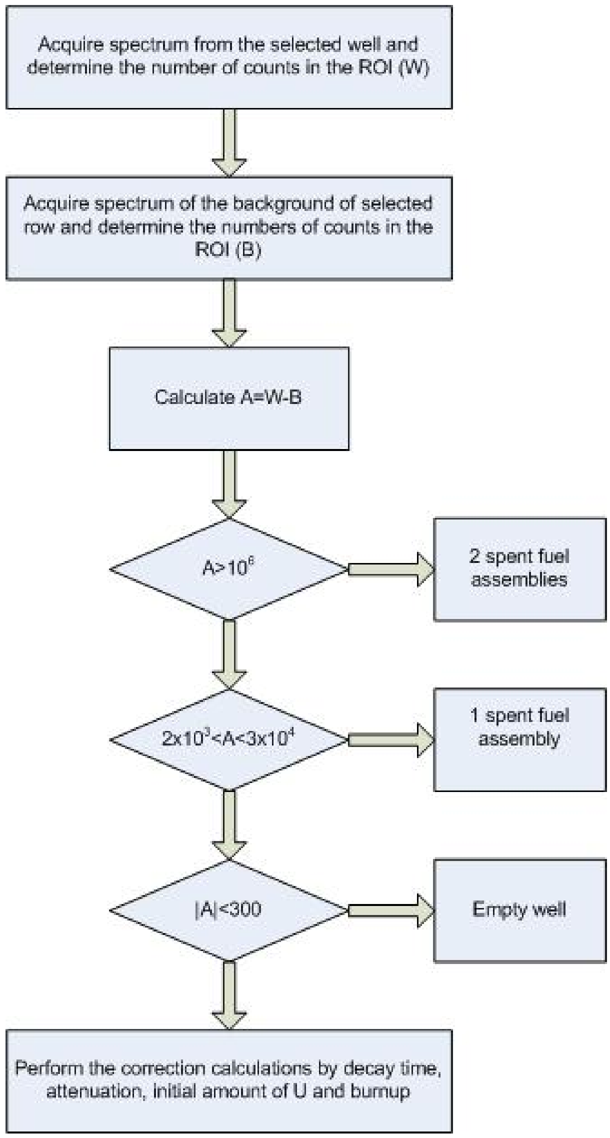

Figure 6: Flowchart of the proposed method.

Although the acquiring spectrum procedure allows avoiding unnecessary exposure to high doses, and ensures that the spectrum belongs only to the contents of the measured well, the spectra taken from different rows are not comparable, because each row has a different background and water level and, therefore, different radiation attenuation. A simple comparison of spectra of different rows would not distinguish between a well with a very decayed fuel assembly and an empty well with a very high background.

To properly evaluate the contents of a well it is necessary to make corrections to the measured activity. Corrections to be considered are the attenuation due to water and lead, the decay time and the mass of Cs-137, which is proportional to the burnup and the initial amount of uranium [2]. When the same detector is used for all measurements it is not necessary to make any correction for detector efficiency, while using always the same collimator permits to ignore the attenuation due to lead thickness. The solid angle correction might even be neglected by assuming that the fuel assemblies are always in the same position, but the other corrections cannot be neglected. For attenuation correction it is only necessary to measure the water level, but the corrections for decay time and burnup require data that are only available in the reactor records and not in the storage facility. To avoid these inconveniences it is proposed to subtract the background from each spectrum. This requires taking a spectrum of an empty well in the same row and subtracting it from the selected well measurement. The background on each row with an acquisition time of 300 seconds, varies from 10^{2} to 2.5×10^{3}. Consequently, we consider the number of counts in the ROI in the well's gamma spectrum and subtract the background in the same ROI from this value.

To standardize the method we fixed the acquisition time in 300 seconds. Typical values obtained are shown in the Table 1.

Material Distribution	**Measured (counts)**	**Measured minus background in the control well (counts)**
2 fuel assemblies	1.2×10^6	1.2×10^6
regular fuel assembly, control fuel assembly	3×10^3 to 3×10^4	2×10^3 to 2.9×10^4
Empty well	Up to 2.5×10^3	-300 to 300

Table 1. Typical values obtained and the correction subtracting the background.

The method allows us to discriminate even between a well with a very decayed fuel assembly and an empty well with high background. In case of an abnormal result, the corrections mentioned above can always be performed.

CONCLUSIONS

This experience showed that it is possible to distinguish among wells containing one, two or no spent fuel assembly. To achieve this goal it is necessary to acquire the spectra of the selected well and the reference empty well of the same row (for background) and determinate the number of counts in the region of interest in each case. After this, we should subtract the background to obtain the absolute activity. Then, by using Table 1 it is possible to conclude if there are two, one or none fuel assemblies in the well.

The different Cs-137 activities observed in some wells, even when they do not compromise the resolution required for this test, can be explained by the attenuation due to the water, the decay time and the particular amount of Cs-137 in the contained fuel assemblies.

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REFERENCES

[1] Safeguards Criteria for Storage Facilities, IAEA, 2007 edition.

[2] Irradiated Fuel Measurements, J. R. Phillips, 18.3.3 *Passive Nondestructive Assay of Nuclear Materials*, Doug Reilly, Norbert Ensslin, Hastings Smith, Jr. and Sarah Kreiner, United States Nuclear Regulatory Commission, Washington, 1991.

Inflammatory Markers of Radiation-Induced Late Effects

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Inflammatory markers of radiation-induced late effects

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INTRODUCTION

Up to now there is no established parameters for the follow-up of delayed radiation injuries . (1)
Late toxicity is generally irreversible and can have devastating effects on quality of life of people exposed either accidentally or during therapeutic radiation treatments .
Histologically, late manifestations of radiation damage include fibrosis, necrosis, atrophy and vascular lesions. Although many etiologies have been suggested regarding these late toxicities, persistent inflammation has been described as playing a key role.
The recruitment of leukocytes from circulating blood is decisive in the inflammatory reaction.. All the steps in the recruitment cascade are orchestrated by cell-adhesion molecules (CAMs) on both leukocytes and endothelial cells, and different subsets of CAMs are responsible for different steps in extravasation (2)
A link between radiation –induced inflammatory processes and alterations in T-cell immunity are still demonstrable in the blood of A-bomb survivors. (3)
The following study was conducted to examine the response of the immune system in the inflammatory reactions in patients with late skin injuries after radiotherapy or interventional fluoroscopy procedures
The expression of adhesion molecules ICAM1 and β1-integrin on granulocytes and lymphocytes , as well as changes in subpopulations of T lymphocytes and the level of C-reactive protein , a well- studied inflammatory marker were evaluated

MATERIALS and METHODS

From 1997 to 2011 over 160 patients were referred to the Radiopathology Committee of Hospital de Quemados del Gobierno de la Ciudad de Buenos Aires (Burn Hospital) for the diagnosis and therapy of Cutaneous Radiation Syndrome. The follow up of twenty one patients that showed late cutaneous reactions graded according to the RTOG / EORTC system is reported here.
The expression of adhesion molecules ICAM1 and β1-integrin was measured by staining whole blood samples with a FITC-conjugated monoclonal antibody mouse anti-human ICAM1 (clone 15.2 ,Chemicon) and a FITC-conjugated monoclonal antibody mouse anti-human INTEGRIN beta 1 CD29 (clone TDM29 ,Chemicon) respectively.
The assessment of T(CD3+), T(CD4+) and T(CD8+)lymphocytes subsets was performed by staining with Tri-Test CD4-CD8-CD3 Reagent (BD) on whole blood sample.
After erythrocyte lysis, the samples were analysed in a flow cytometer (BD FACSCalibur) using CellQuestPro software.
The level of CRP was measured on plasma samples with an immunoturbidimetric assay (Full Range CRP, RANDOX)

RESULTS and DISCUSSION

Characteristics of the study patients

Median age (ranges) : 63 (49-79)
Late effect is considered after three month of the radiation procedure.
The distribution by etiology is representative of all patients referred to the Burn Hospital during the period 1997-2011 (Fig 1)

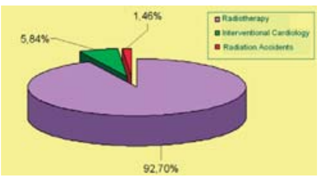

Fig 1 . Distribution by etiology of patients treated at Burns Hospital, period 1997-2011, N= 160

Late toxicity was evaluted according to the use of the RTOG/EORTC Late Radiation Morbidity. Tissue: skin

Grade 1: SkinSlight atrophy;Pigmentation change; some hair loss

Grade 2: Patch atrophy; moderate telangiectasia; total hair loss

Grade 3: Marked atrophy; gross telangiectasia

Grade 4: Ulceration

Cases of skin injury

Panel A
Patient who underwent Rt for ovarian cancer during mid 70s. Cyclical evolution with exacerbation crisis from 2000 up to now

Panel B
Rt for Thymoma in 1984. The patient presents at Burns Hospital in 2010 complaying from back pain and later ulceration since 2008.

Panel C
Radiaton injury following interventional cardiology procedure reported as difficult, requiring prolonged duration of fluoroscopy.

The analysis of adhesion molecules expression revealed a higher expression of β1 Integrin on gated lymphocytes of Grade IV patients compared to non exposed controls (Fig 2). It was also noted a decrease in its expression values in the follow up of patients with good response to therapeutic treatment (Fig 3).

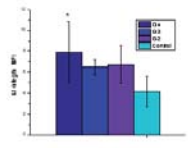

Fig 2. β1 Integrin as Mean Fluorescence Index (MFI) on gated lymphocytes of patients graded according to RTOG/EORTC System. * p< 0.05 compared to control

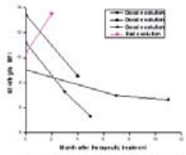

Fig 3. Changes in β1 Integrin expression on gated lymphocytes of some patients as response to medical treatment

Three-color immunofluorescence flow cytometry of lymphocyte subsets showed a tendency to a decrease in the T(CD4+) / T(CD8+) ratio of G4 patients with bad evolution compared to G4 patients with good evolution (Fig 4). The frequency distortion of thymic precursors CD4-CD8- (Double negative DN) and CD4+CD8+ (Double positive DP) in peripheral blood observed in G4 patients, could be suggesting disturbance in T-Cell homeostasis (Fig 5).

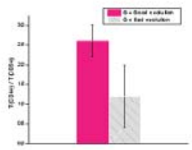

Fig 4. T(CD4+) / T(CD8+) in G4 patients with good and bad evolution

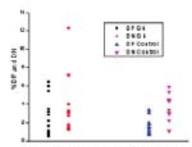

Fig 5. Percentage of Double Positive and Double Negative thymic precursors in G4 patients and control sample

The level of C Reactive Protein (CRP) , a widely used inflammatory marker, showed higher values in patients in acute phase (52,1±47.4mg/L*) and patients with late toxicity but in exacerbation crisis (13.5± 5.3mg/L*) with respect to patients with late radiation injury (1.9± 1.4mg/L). * p< 0.01

The parameters analysed, which require confirmation in a larger study , in combination with other inflammatory indicators, could be used as potential follow-up markers of the chronic radio-induced inflammation process just as its response to therapeutic treatments.

REFERENCES
(1) Scherthan H et al. Rad Res 167 :615, 2007
(2) Ulbricht H et al. Trends Pharmacol Sci. 24:640, 2003
(3) Kusunoki Y et al. Rad Res 174 :870, 2010

State and Regional Systems of Accounting for and Control of Nuclear Materials Cooperation between International, Regional and States Safeguards Organizations: an Evolving Issue

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STATE AND REGIONAL SYSTEMS OF ACCOUNTING FOR AND CONTROL OF NUCLEAR MATERIALS COOPERATION BETWEEN INTERNATIONAL, REGIONAL AND STATES SAFEGUARDS ORGANIZATIONS: AN EVOLVING ISSUE

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ABSTRACT

Cooperation between the IAEA[1], States and regional organizations is increasingly important to ensure effective accountancy and control of nuclear material in peaceful uses. The IAEA, SAGSI[2] and institutions such INMM[3] and ESARDA[4] have recognized the relevance and the evolving role that SSAC[5] and regional organizations play to this aim. In this context, it is important to take steps to ensure the effectiveness of the system and the optimal level of relationship between these organizations so as to maximize the benefits for each party, particularly in those cases where well developed systems exist.

Moreover, expansion of nuclear energy requires concerted efforts towards building competence in safeguards in all relevant States. This is also important with respect to other aspects of non-proliferation. In this scenario there is agreement on the need to have effective state organizations that fulfill international safeguards and other security obligations. However, the roles and duties of SSAC and the possible scope of cooperation between the IAEA and SSAC are still under evolution. This paper discusses possible ways and means to build competence in safeguards and how the international community could be more proactive in establishing a framework including the various dimensions of the cooperation in safeguards and other security matters between all parties concerned. The establishment of a forum and a network of interested parties under the auspice of interested organizations could be one mechanism to exchange best practices and experiences.

INTRODUCTION TO CAPACITY BUILDING IN SAFEGUARDS IN ALL STATES

This paper is part of a series of essays aimed at highlighting the role of SSAC and regional organizations to ensure that the safeguards' goals are accomplished at higher standards. Previous works stressed the relevance of SSAC and regional safeguards arrangements like ABACC[6] to nuclear non-proliferation and international security as well as the role of all parties in furthering cooperation to address safeguards objectives fully, including the notion that the IAEA must use the 'findings' of such entities in lieu of its own. While elaborating further on the role of cooperation in modern safeguards, this piece focuses more in the need to build competence in

[1] International Atomic Energy Agency
[2] IAEA´s Standing Advisory Group on Safeguards Implementation
[3] Institute of Nuclear Materials Management
[4] European Safeguards Research and Development Association
[5] State Systems of Accounting for and Control of Nuclear Materials
[6] Brazilian-Argentine Agency for Accounting and Control of Nuclear Materials

safeguards in all States, regardless of their status with respect to nuclear activities. It also discusses possible mechanisms that the international community, through a new approach to cooperation, may pursue to ensure such competence.

FACTORS INFLUENCING THE INCREASING ROLE OF SSAC

It is evident that technically competent SSAC are increasingly important to ensure effective and efficient international safeguards and to contribute to other important state functions such as attaining a good level of protection of nuclear material and activities. One factor that is influencing this trend is the perception that a high level of safeguards globally depends on the existence in any state of legislation and regulations that contribute to ensure that safeguards are being implemented properly. Another relevant factor is the increasing awareness on the complementary benefit of safeguards, safety and security *("3s" approach).* The benefit to develop larger and more comprehensive states authorities empowered with a legal framework, competence and tools to ensure compliance in the "*3s approach*" is being considered in countries that have recently embarked on nuclear power and other states with advanced nuclear fuel cycles.

Besides, the impact of the relationship between safeguards and security is not trivial from the side of the states having to respond to new security challenges. Discussions at certain meetings about SSAC are taking place to consider a broader role of the states organizations to address new international requirements and recommendations on the non proliferation and security areas like the requirements set up in the UNSC 1540 Resolution.

Thus, it is plausible to say that a driving force to the increasing role of SSAC and regional endeavours is the recognition that the framework and the international scenario where nuclear energy used to take place has drastically changed. Safeguards are becoming more complex; nuclear cooperation for peaceful applications has grown in a globalized world, there are new challenges due to the expansion of nuclear energy. The breaches to non-proliferation and the backdrops or lack of tangible progress in nuclear disarmament are also affecting the environment of nuclear energy developments.

All these reasons explain why there is a need to make further efforts towards increasing the level of awareness and knowledge in safeguards at the level of all states for them to respond to this complexity.

CAPACITY BUILDING IN SAFEGUARDS – STATES WITH SMALL OR WITHOUT NUCLEAR ACTIVITIES: A DIFFERENT PERSPECTIVE

While certain progress has been achieved, the Safeguards Implementation Report (SIR) continues to report that there are still States that have to sign a CSA[7] and many states don't have yet a SSAC or a government authority in place to fulfil safeguards basic requirements. The need of legislation and infrastructure in states are in many instances absent or insufficient.

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[7] Comprehensive Safeguards Agreement.

Nowadays, that a state should have a minimum level of competence in safeguards is important regardless of the existence of nuclear activities. This may sound awkward for states where the least of the problems are far from being related to nuclear energy, but the transformation of the international nuclear environment and the trend in safeguards towards their implementation at the state-level concept show the increasing need for that. However, it is valid also to say that the fulfilment of the requirements foreseen in the CSA or other safeguards agreements should be clearly commensurate to the "nuclear reality" of such states (e.g., States without nuclear activities and no plans to initiate them in the foreseeable future). There is a need for further pursuing a graded and more pragmatic approach and to start the capacity building process by the establishment of a set of guidelines focused on central issues with simplified yet adequate safeguards procedures and good practices. SSAC in a country without nuclear activities should not be expected to perform all requirements noted in the safeguards agreements, but should rather be well aware of them and more importantly of the provisions related to relevant notification and reporting obligations.

This approach would favour the possibility to optimize limited resources in these countries by establishing procedures that have a greater chance to be fulfilled and by combining duties that can be relatively linked (For example: It could be satisfactory to start training in safeguards important issues for a person that works in a health government organization dealing with a radiation source, etc.).

The action plan implemented by the IAEA is obtaining good results, but more needs to be done to improve the situation. The IAEA should simplify safeguards requirements as much as possible for States without nuclear activities and should provide a simple and very handy guide that would be easy to follow.

In addition, concerted efforts from regional organizations and states' networks, NGOs and forums like INMM and ESARDA could be made for reaching those states and to cooperate in capacity building in safeguards through informative workshops, the sharing of good practices, the elaboration of simple legislation and regulatory requirements, and the identification of a candidate institution to act as a SSAC or an equivalent concept. Priority should be given first to those states without a safeguards agreement and/or still having the old version of the SQP in place. The new comers with safeguards agreements and updated SPQ would also enjoy the same priority.

CAPACITY BUILDING IN SAFEGUARDS – STATES WITH NUCLEAR ACTIVITIES AND CAPABLE SSAC (WITH OR WITHOUT A REGIONAL SAFEGUARDS SCHEME)

As it was noted earlier, safeguards effectiveness and efficiency are based on the existence of competent SSAC. A corresponding priority should be given to develop new methods, technology and concepts to increase the cooperation between the IAEA, SSAC and regional organizations (when they are in place) in countries where a well-established safeguards infrastructure exists. In this case, the IAEA must consider ways and means to make full use of the verification activities carried out by states and regional arrangements. The existence of a regional organization that also implements safeguards constitutes an added value to safeguards goals and more generally to non proliferation that allows expanding the cooperation further. Increasing the role of SSAC or regional schemes in a way that the IAEA can use their results more fully could only be achieved if there is a change in the current approach to cooperation.

SOME IDEAS TO ACHIEVE SUSTAINABILITY IN NATIONAL SAFEGUARDS CAPACITY IN EACH STATE

There are initiatives regions like the Asia-Pacific Safeguards Network aimed at promoting safeguards good practices and exchange of lessons learnt. These endeavours should be expanded to other regions through the establishment of similar organizations or through the identification of existing bodies that could play a similar role.

A peer review mechanism where states could share information on SSAC good practices and lessons learnt could also be an option to consider. In such case, there is a need to agree on basic principles and practices that any SSAC has to address. This can take the form of a code of conduct or even a safeguards convention similar to the nuclear safety one. Despite the benefit of such mechanisms, a code or a convention and even a peer review system in an asymmetric environment (i.e., existence of nuclear weapons states) might not be realistic, unless they are developed and implemented by CSA states and then open to the participation of others under certain specific framework.

Safeguards are being subject to substantive changes both technologically and conceptually. As noted before, this more complex scenario justifies the increasing importance of SSAC. Therefore, the IAEA has also to re-visit its approach to cooperation with states to ensure that its efforts are adequate to help them in establishing and maintaining good SSAC. The IAEA should investigate and promote efforts in developing concepts and techniques to share its verification capabilities with states. In cases where regional organizations that already partner the IAEA, they should play a key role in designing the optimal frame and extent for that cooperation.

The suggestion in this area is that the IAEA redirects or expands its cooperation to develop a new partnership approach that includes the sharing of technology for better states' safeguards and to sustain robust SSAC. The sharing of equipment and other tools as well as the cooperation to develop and maintain NDA[8] and DA[9] capabilities for safeguards purposes in the states could be used by the IAEA and the states (and regional organizations where they exist). This approach to cooperation would lead to a win-win outcome. The further discussions on the potential of this new approach to technology for safeguards can take place in forums like the IAEA's MSSP[10], the Next Generation Safeguards Initiative and the further development of the concept known as 'safeguards by design'. The IAEA could also promote and coordinate sates-to-states cooperation (bilateral, multi-bilateral, etc.) in specific regions of interest. All can be benefited by considering technologies that can be shared in support of better safeguards through building technical

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[8] Non-destructive Assays
[9] Destructive Assays
[10]IAEA's Member States Safeguards Support Programme.

capacity in the states systems. The IAEA's R&D Programme for Verification includes a project in which one of the goals is to cooperate to promote good SSAC in place through training. This is an area that can be expanded or modified to include the application of this recommended approach.

The role of INMM, ESARDA and other similar organizations is important in promoting the development of new concepts and techniques that help the international community to maintain a good standard of safeguards worldwide. Moreover, they are already engaged in addressing the need of having robust SSAC and in discussing the cooperation facet of safeguards as a fundamental precondition to ensure their effectiveness. Maintaining and ideally, expanding special sessions like the one where this paper is presented is just one example of the activities these organizations could lead or promote. Institutions like INMM could not only serve as a forum to discuss this relevant topic, but could also include side events in the margins of the ISD[11] devoted to the exchange of good practices and lessons learnt among SSAC, regional organizations, networks associations and the IAEA.

THE NEED TO ENSURE ADEQUATE STAFFING AND THE ROLE OF TRAINING AND EDUCATION

Needless to say that skilled and highly-technically qualified staff is 'the' turning point to build competence in safeguards in states, the IAEA and regional organizations. Adequate personnel are crucial to ensure effective safeguards implementation at all levels. There is a gap to address in this field. The lack of qualified people is more or less general to all nuclear disciplines, so there is a need to continue efforts towards promoting incentives and capabilities to offer educational and career opportunities to young people in all states and especially in developing countries.

From the side of the states, it is important that legislation and regulations recognize the role of the SSAC staff; both at the level of the state authority and at the level of the operator, education and training needs and programmes are identified and are in place. Safeguards are technically and multidisciplinary based on nuclear and other hard sciences. The regulatory body, when it exists, or the national point of contact for IAEA technical cooperation, should take a leading role in promoting that a career in safeguards be a matter of reward and prestige for those professionals and technicians that chose to work in this field. As part of the nuclear regulatory requirement, one possibility to consider is that the safeguards officer position at a nuclear facility would be part of the license process, and if the SSAC's duties are entrusted to the regulatory body, it should take measures to ensure a career of good prospect. In addition of having qualified staff, international safeguards need also to consider other features such as geographic distribution and participation in all positions and duties. In this regard, it is important that the IAEA keeps a constant look into this matter to ensure equal opportunities to developing countries and for all geographic regions. It could be positive that the IAEA exercises a more proactive role in promoting a safeguards career in member states. In the end, IAEA's people are recruited from them.

[11] INMM International Safeguards Division

In conclusion, to fill the gap and to get high qualified people, knowledge management and human resources programmes should engage all parties concerned in a plan to promote training and education in safeguards as well as in providing a prospect of a good career. The IAEA could take a lead in designing and implementing this plan as part of the cooperation to ensure robust SSAC.

OTHER REQUIREMENTS OF THE SSAC: LEGISLATION, ECONOMIC RESOURCES, INDEPENDENCY

States should establish legislation and regulations to ensure that a SSAC or an equivalent concept is in place to fulfil basic safeguards requirements. A SSAC should not only have adequate staffing and technology to comply with its mission, but also sufficient economic resources and independence from the operators. Legislation should also address the provision of funding the state's safeguards duties. This paper is not aimed at discussing in detail this and other topics related to SSAC's competence, but to highlight areas where a fresh look into international cooperation could contribute to having good SSAC.

COOPERATION AS A WHOLE: A CONCEPT TO BE REVIEWED AND REVITALIZED AS A BASIS FOR SAFEGUARDS IMPLEMENTATION

Cooperation in Safeguards continues to be their basis and their foundation. It is for all parties the obligation to cooperate. Nowadays, a limited view that considers only the cooperation that is due from the side of states reduces the chances to optimize and expand safeguards effectiveness and efficiency.

We need to progress towards a new partnership approach in which all parties have a clear role in implementing meaningful safeguards and each party can enjoy an optimal combination of the tools and techniques available to ensure good accountancy and control of all nuclear materials in all nuclear activities. A drastic change in the safeguards culture is urgently needed. We have to build safeguards cooperation in a different direction from that of today: modern safeguards require a team-oriented and partnership approach that not only optimizes the use of existing resources but more importantly creates an environment of confidence (e.g., inspectors working together, the establishment of agreed procedures to jointly use safeguards techniques and equipment). The IAEA has a key role to play to achieve this approach fully.

Current initiatives (for example the next generation safeguards initiative) could also address this concept by including features that allow all parties to take advantage of safeguards developments. The IAEA should review its approach to cooperation with States and Regional authorities on the basis of this fresh approach.

FINAL REMARKS

Nowadays a SSAC or a state having a minimum level of competence in safeguards is important regardless of the existence of nuclear activities. However, flexibility and the notion that 'no one size fits all' are important concepts to bear in mind at establishing SSAC requirements.

The cooperation of the IAEA to assist states in building competence in safeguards should be expanded beyond current activities. Although the IAEA has increased the number of training courses, workshops and seminars to collaborate with states in ensuring the establishment and maintenance of good SSAC, a fresh look into the scope and nature of the cooperation is needed. Appraisal missions, although useful to assess SSAC competences and to indentify good practices and recommendations, do not seem to be enough to build effective SSAC worldwide at the required pace.

In modern safeguards, not only the existence of scarce resources leads us in the direction of sharing technical and human resources capabilities, but more importantly, towards a high level of safeguards competence in all states in a sustainable manner. International cooperation should consider other means and ways to promote competence in safeguards and basic SSAC in all states. There is an urgent need to revisit and change the IAEA's approach to cooperation to a new team-oriented approach.

The concept of "partners looking forward to achieving the same shared safeguards values and goals" should prevail.

Exchange of experiences and the sharing of good practices can be further encouraged by the IAEA, regional organizations and other interested parties like INMM, ESARDA. The IAEA should also consider expanding the MSSP project related to SSAC so as to increase cooperation in other areas like sharing technology and helping states to install and maintain technical capabilities in safeguards.

The increasing importance that SSAC and regional organizations have in international safeguards is clear. Cooperation continues to be central for effective safeguards. The complexity of IAEA safeguards requires a high standard of competence in states safeguards. However, further efforts are needed to address the fact that still many states don't have a SSAC. A simplified and graded approach should be further investigated and established for states without nuclear activities.

Remote Monitoring in Safeguards: Security of Information and Enhanced Cooperation

Galdoz, E.; Calzetta, O.; Fernández Moreno, S.; Llacer, C.; Díaz, G.D.; Vigile, S. and Brunhuber, C.

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REMOTE MONITORING IN SAFEGUARDS: SECURITY OF INFORMATION AND ENHANCED COOPERATION

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ABSTRACT

Unattended systems with remote transmission capabilities (RM) have the potential to improve safeguards efficiency. Moreover, the evolution of technology and the steady growing of nuclear materials subject to control, lead modern safeguards increasingly utilizing unattended equipment with the capability to store relevant data for long periods of time coupled with the option of being remotely accessed and checked. Remote inspection is still a concept under development, but it may end to be a powerful more efficient verification modality in medium term future. An important part of drawing meaningful safeguards conclusions rests on authenticity and reliability of the information on nuclear material and facilities acquired through the various verification activities and measures applied by IAEA and regional safeguards organizations, like ABACC.

The increasing utilization of such technology to further optimize safeguards responds to a multifaceted environment where security of information for all relevant parties is of utmost importance. From the point of view of the IAEA and ABACC, the use of any technology for safeguards application, and specially the use of RM, requires to ensure the security of data collected to guarantee the validity and veracity of such information throughout the whole process (e.g., from collecting to reviewing). This is also valid to the SSAC involved in the process. Information security is also relevant for States and Operators. Assurance should be given that the information could not be withdrawn by non-authorized entities and that facility data is also fully secured. Another important aspect related to RM that may also fall in the security aspect of safeguards relevant information that merits further consideration, is the sharing of information between organizations like ABACC and the IAEA as well as the possibility to make this data available for States authorities purposes.

This paper discusses three main themes related to RM: (i) the extent to which security is key for RM application acceptance and use for the IAEA, ABACC, States and Operators, ii) the sharing of relevant safeguards data for all the parties concerned, iii) a scheme agreed between ABACC and ARN to trial a RM transmission and a possible approach for international safeguards application.

REMOTE MONITORING APPROACH

Safeguards implementation is essentially a technically driven process, in which large amounts of data, sometimes of radically different nature, must be collected (i.e. nuclear or non-nuclear measurements, surveillance images, conventional or electronic support documents), verified for safeguards purposes (using specific authentication techniques), stored (in different storage media like magnetic tapes, memory cards, removable hard drives, optical disks), secured (using encryption and/or authentication techniques), transported (carried by the inspector, or transmitted over public communication lines) and finally analyzed (by data comparison, measurements evaluation, images review, etc.) using different techniques and grades of precision, depending on the specific facility approach. All of these, for a large and increasing number of facilities around the globe. This brief summary seems to show heterogeneous, complex and cumbersome data management scenarios. And so it is.

Increasing efforts are continuously made to introduce new technology in order to reduce or minimize the complexity and costs involved in this process, thus increasing the efficiency without compromising the effectiveness, precision and completeness of the verifications performed.

One of the important efforts consists of the transmission of the information collected by communication capable devices running at the monitored facilities to servers located at the monitoring agencies headquarters, where inspectors can analyze the data in detail using software tools aiding the job, and gather conclusions in a more efficient and less intrusive way. This is the basic idea of Remote Monitoring Systems, in which unattended monitoring equipment, such as optical surveillance systems, are connected to a central storage system using any available communications link, to transmit the collected information. The information must be properly conditioned prior to be transmitted over publicly accessible lines, by encrypting and digitally signing the data packets, as described later.

SECURE TRANSMISSION: DIFFERENT PERSPECTIVES AND REQUIREMENTS

Several security requirements must be met when any piece of information is to be transmitted over a publicly accessible line, to ensure the data shall reach its intended destination unaltered, the originator identity can be authenticated, and no access to the information by unauthorized third parties is possible while transmitted through public lines.

In the general case of Remote Monitoring Systems for safeguards application, the data is originated in equipment under control of an organization intended to receive and process it in a different place, that is, the monitoring agency. Authentication techniques should be applied immediately after the data is produced to reduce the risk of data tampering. When two organizations or more share the equipment and its data under certain agreed procedures for joint use, they have also to agree on how to apply and fulfill the above-mentioned requirements.

To securely transmit the information to the final destination, two different approaches can be considered. The first is based on the use of a private communication line. This approach is nowadays still expensive and difficult to implement. The second approach uses normal communication lines like telephone lines or internet links, as a physical layer. The information must be encapsulated using strong encryption and authentication techniques, widely available in reliable and inexpensive VPN (Virtual Private Network) implementations. Using this approach, even network infrastructure owned by the Facility Operator (FO) or the National Authority (NA) can be employed, not compromising the system trust.

Not only should the perspective of the monitoring agency be taken into consideration, but normally the FO and/or the NA have also conditions that must be met in order to accept the RM approach. The NA is required to protect the information taken from nuclear installations for security, industrial and technological aspects, among other reasons. In this regard, the NA should be able to ensure and demonstrate that the information transmitted is that agreed upon, and also that the equipment cannot be remotely accessed in order to modify parameters without prior agreement. Therefore, to implement a RM application it is also important to satisfy the State/Operator security requirements.

SHARING OF DATA: ANOTHER IMPORTANT CONSIDERATION

Successful safeguards implementation depends on the cooperation of all parties concerned. In the case of a RM application, that implies the IAEA and ABACC, the NA and the FO.

The existence of a robust and effective SSAC empowered with the necessary legal authority, which is independent from operators, and has adequate resources and technical capabilities to administer the requirements of safeguards agreements to properly verify nuclear material accountancy and control systems at nuclear facilities and LOFs is of utmost importance.

All involved Agencies (i.e. IAEA, ABACC, and the NA) should discuss and agree procedures to allow the sharing of data to the fullest extent possible. This criterion should not compromise a relevant diversion scenario.

Simultaneous full sharing of the information transmitted with the NA may be considered as difficult to accept for the Agencies. However, there is the possibility of sharing data in a way that would not compromise the safeguards approach. For example, let us consider the situation when the amount of data and due time are such that a surveillance malfunction or potential flaw, relevant for the approach, can be disclosed without negative impact on the evaluation. This is a scheme that would allow all parties to obtain the fullest possible benefits of RM applications.

JOINT REMOTE MONITORING BETWEEN IAEA AND ABACC

A more technically complex situation involves multiple agencies jointly monitoring several facilities, and that is the case of ABACC-IAEA joint safeguards. As noted above, in this case both agencies share the information collected by unattended surveillance equipment, owned by one of the agencies, but provisions must be ensured so both agencies can obtain independent conclusions.

Once the information is acquired and conditioned, it is simultaneously transmitted to the agencies´ headquarters for further storage, analysis and review. Parallel VPN channels should be used for secure transmission.

Another feature to be considered is that corrective or preventive maintenance tasks must be done when some malfunction is detected. In some occasions this can be done remotely. In this situation there are requirements from both the NA and the agency which does not own the equipment that need to be adequately addressed. One condition to be met is that even when the owner agency is responsible for accessing the system for maintenance purposes, the other parties involved must be aware of the access, and able to audit the access to avoid undesired or not agreed changes that could affect the system operation and/or performance.

A VARIATION: STATE OF HEALTH REMOTE MONITORING

Depending on the nature of the facility under monitoring, the State can require safeguarding industrial, commercial or national security related aspects to the extent that the information gathered by the surveillance systems is not to be retrieved outside the facility. All review activities must be performed during the inspection time, inside the facility boundaries, and the conclusion must be obtained prior to leave the installation.

Even if that is the case, remote transmission of relevant information indicating the State of Health (SoH) of the equipment and components, excluding any sensitive information, can significantly improve the overall system performance and efficiency, by allowing prompt alerts when malfunctions occur, or symptoms announcing such malfunctions are detected. As a result, safeguards intrusion and inspection effort can be reduced.

The security considerations are essentially the same as in the full Remote Monitoring implementation, but the State normally may require extra auditing capability over the transmitted information.

SoH DEMONSTRATION TEST BETWEEN ABACC AND IAEA

A demonstration experimental set-up is being tested between the ABACC and the IAEA, where the SoH information gathered from an SDIS surveillance system (owned by and located in ABACC HQ) and two NGSS cameras (owned by IAEA, also located in ABACC HQ) is simultaneously transmitted to both agencies storage sites for further analysis. Symmetrical remote access to the monitored surveillance system for maintenance purposes is also granted. By symmetrical access must be understood that no agency can remotely access the system without effective acknowledge of the other, and all activities can be monitored in real time, as explained later. All communications that use public Internet services are encapsulated inside VPN tunnels.

The traffic is managed by the firewall rules set-up in the VPN devices participating in this multi-tier communication scheme. The VPN devices selected for this project are all qualified by the IAEA for remote transmission of safeguards data. The applied VPN rules define which data paths are permitted, in which direction, and what services (i.e Transmission Control and Internet Protocols, TCP/IP services) are expected to be used. All other traffic is denied by default. In this way, it is assured that all data are sent to the intended destination servers only, and only the authorized computer, with the proper temporary permits granted, can access the surveillance system. Access to the VPN devices configuration is controlled by passwords shared by both agencies, ensuring that changes can only be done jointly.

Figure 1 shows a scheme of the network configuration between the simulated facility and both agencies sites. The surveillance system (A) is connected to Internet by using a dedicated ADSL link, simulating the connection type normally available at facilities. The Joint RM Server (B) is connected to the Internet using ABACC regular Internet provider, and identified by a public IP address. The ABACC RM server and Joint RM Workstation (C) are also connected through the same provider and using a different public IP address; and finally, the IAEA RM server (D) is also connected to Internet using regular Internet resources at IAEA HQ. In this way, a realistic communications scenario is fully simulated. This Joint Use Server approach has been successfully in use for more than three years between IAEA and EURATOM.

Only the computer labeled as Joint RMS Workstation is allowed to initiate a Remote Desktop session at the monitored SDIS server and that access is controlled by the IP Tracking VPN device (See details in Annex 1). In that way, a session can be initiated only if the access is remotely granted from within IAEA HQ. In the same way, IAEA can only start a session from within that workstation, and the access is granted by ABACC simply by turning the Workstation on. This scheme allows a completely symmetric access right, and all maintenance activities must be carried out jointly, as requested by design.

The surveillance system (A) also includes an extra computer identified as Traffic Recording Device, which works as a proxy between the monitored systems and the rest of the network. This proxy computer runs ad-hoc software developed by IAEA, which intercepts all traffic coming from and going to the monitored system, and all that traffic is recorded for auditing purposes prior to be dispatched to the intended destination. These auditing records are stored on the recorder hard drive, and can be requested by the FO and/or the NA at inspection time to verify that only the agreed information was transmitted during a previous period under review.

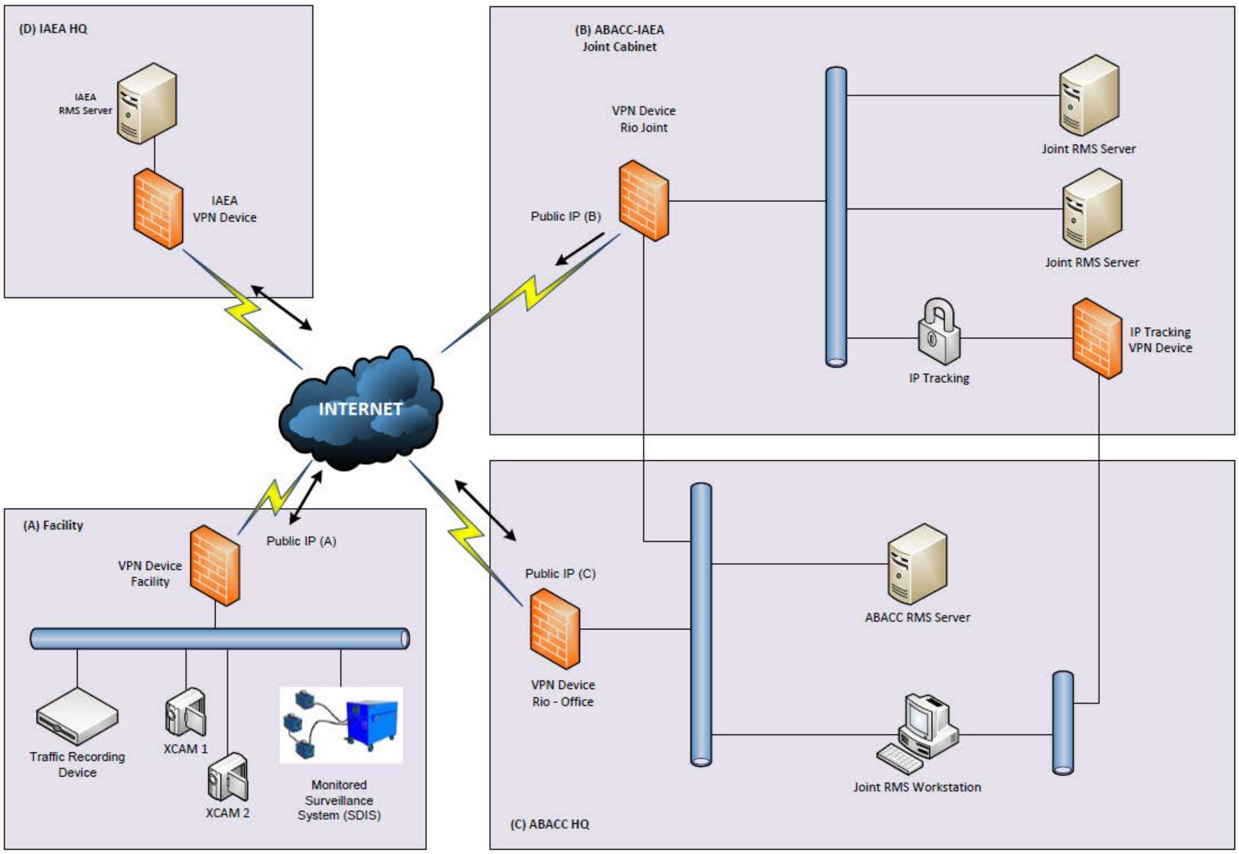

Figure 1: Network diagram of the Joint SoH demonstration pilot.

SoH TRANSMISSION FROM THE PERSPECTIVE OF THE NA

The VPN approach proposed in this paper makes the use of network infrastructure provided by the National Authority (NA) possible. Therefore it is a preferable option as the NA is responsible for the access to the facility since the use of the FO/NA network does not compromise the reliability of the system and fulfills the other Monitoring Agencies requirements. This would also allow the NA to keep a log of transactions time, direction and amount, which would be used to verify at accorded time that the agreed data was effectively transmitted and that all data flowed outward and not inward.

It is necessary to keep in mind that due to security concerns, encrypted data is usually not allowed to flow through the FO/NA network. In order to route this encrypted data some rules should be enforced on the equipment. To achieve these data paths, TCP/IP services and the data flow intended direction should be agreed with the NA.

To complement this, as in the ABACC-IAEA test previously mentioned the surveillance system (A) also includes an extra computer identified as Traffic Recording Device, as explained above.

OTHER VARIATION: FUTURE IMPLEMENTATION AND UPGRADE

In the future, the development of a RM scheme capable of corrective or preventive maintenance tasks could be forethought. In some occasions these corrective or preventive maintenance tasks could be done remotely. In these cases, there are requirements from both the NA and the agency that does not own the equipment that need to be adequately addressed. One condition to be met is that even when the owner agency is responsible for accessing the system for maintenance purposes, the other parties involved must be aware of the access, and able to audit the access to avoid undesired or not agreed changes that could affect the system operation and performance. Also, as each and anyone of these accesses may affect the RM system performance and/or the collected data integrity, they should be previously agreed with the NA. For the same reason, all incoming data flow should be logged for eventual review.

In the Figure 1, with public IP (A) provided by the NA, only the computer labeled as Joint RMS Workstation is allowed to initiate a Remote Desktop session at the monitored SDIS server and that access is controlled by the IP Tracking VPN device. In that way, a session can be initiated only if the access is remotely granted from within IAEA HQ. In the same way, IAEA can only start a session from within that workstation, and the access is granted by ABACC simply by turning the Workstation on. According to the scheme proposed, and as the NA supervises the connection to the facility, a completely symmetric access is granted, and all maintenance activities must be carried out jointly, as requested by design.

Some tools to store and replay the Remote Desktop sessions performed by the agencies with maintenance purposes have also been successfully tested. Such tools allow the auditing authority to easily verify the tasks carried out on the surveillance server should such sessions have occurred during the surveillance period under review.

CONCLUSIONS

The preliminary results of this demonstration test involving a multi-agencies remote monitoring system with delayed auditing capability show that the system as depicted is reliable, and all design requirements of security and access permits granting are fulfilled, assuring that all parties can verify the proper use of the system, without compromising the reliability and data confidentiality. The ad-hoc proxy software involved is simple enough, and the security and recording phases are implemented using commercial off the shelf software that can be independently assessed by any of the parties involved in the project.

ANNEX 1: WHAT IS "IP TRACKING"?

The Juniper VPN device has a special feature called "Interface Failover with IP Tracking". The manual tells us: "You can specify that when certain IP addresses become unreachable through the primary Untrust zone interface, the security device fails over to the backup Untrust zone interface even if the physical link is still active. ScreenOS uses Layer 3 path monitoring, or IP tracking to monitor IP addresses through the primary interface. If the IP addresses become unreachable through the primary Untrust zone interface, the security device considers the interface to be down, and all routes associated with that interface are deactivated. When the primary Untrust zone interface changes to the down state, failover to the backup Untrust zone interface occurs."

We use this feature for the following approach: The IP Tracking device (access device) located in the sealed cabinet is the only network path into the sealed cabinet. The device has two Untrust interfaces. The second one is not connected. The primary interface is connected to the main VPN device in the cabinet which maintains tunnels to the facilities and to IAEA and ABACC. The access device tracks the IP address at the remote endpoint in IAEA HQ. The IP packets are traveling from the access device via the main VPN device through the tunnel to Vienna. By not responding to the ping, the access into the cabinet can be cut from Vienna.

ABACC has direct access to the access device, so ABACC must not have the ability to change the access device configuration. In other words ABACC must not have the ability to switch off IP Tracking.

Implementation of Safety and Security Issues in the Transport of Radioactive Material in Argentina

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IMPLEMENTATION OF SAFETY AND SECURITY ISSUES IN THE TRANSPORT OF RADIOACTIVE MATERIAL IN ARGENTINA

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Abstract. This paper is intended to describe implementation of safety and security issues in the transport of radioactive material by the Nuclear Regulatory Authority (in Spanish Autoridad Regulatoria Nuclear, ARN), which is the Competent Authority of Argentina in Safety, Security and Safeguards of radioactive and nuclear material. There are depicted main regulatory activities dealing with the mentioned issues, and relevant milestones of national regulatory standards and guidance applied, that are based on requirements and guides from IAEA. Interfaces between Safety and Security sections are most of the times complementary but sometimes conflictive, therefore the resolution of such conflicts and goals achieved during their implementation are also commented; as well as future joint planned activities between both sections of ARN as a way to provide safety and security without compromising one or the other.

1. Introduction

This paper aims to describe the main aspects in the regulatory implementation of Safety and Security issues related to the transport of radioactive material by the Nuclear Regulatory Authority (in Spanish, Autoridad Regulatoria Nuclear, ARN), which is the Competent Authority for regulating transport of radioactive material (RAM) in Argentina.

As early as 1946, the first activities related to nuclear technology began in Argentina. They were institutionalized and consolidated on May 31, 1950, with the creation of the National Atomic Energy Agency (in Spanish, Dirección Nacional de Energía Atómica), later renamed National Atomic Energy Commission (in Spanish, Comisión Nacional de Energía Atómica, CNEA), by Decree No. 10936 [1]. Since then, this Commission has devoted to the study, development and application in all aspects related to the peaceful use of nuclear energy, becoming nowadays the promoter agency in this field in the country.

Two of the first Argentine persons involved in safety issues both domestically and internationally, including transport, were Dan J. Beninson and Alfredo L. Biaggio.

In 1994, the Nuclear Regulatory Agency (in Spanish, Ente Nacional Regulador Nuclear, ENREN) was established by Decree No. 1540 [2], with the goal to supervise and regulate the nuclear activity, previously in charge of CNEA.

Few years later, ARN was established as an autonomous body reporting to the President of Argentina by Act 24804, known as the National Nuclear Activity Act [3], which came into force on April 25, 1997, replacing the ENREN. ARN is empowered to regulate and control the nuclear activity with regard to radiation and nuclear safety, physical protection and nuclear non-proliferation issues. Its objective is to establish, develop and enforce a regulatory system applicable to all activities involving radioactive material carried out in Argentina. In addition, ARN is the advisor of the Executive on issues under its purview. The mentioned law was implemented by the Regulatory Decree No. 1390 of 1998 [4].

2. Legal framework

An effective legal framework is essential to ensure and facilitate safe and secure transport of RAM. Domestic legislation and international recommendations have an active role strengthening long-term control over the transport of RAM, and are periodically reviewed to ensure they remain effective.

The Argentine regulatory standards are performance oriented: they are not prescriptive but define compliance with safety objectives. How these objectives are achieved is based on appropriate decision made by the organization that deals with the pertinent facility licensee or transport approval certificate. The organization must demonstrate to the Competent Authority or Regulatory Body, depending on is regarding to safety or security respectively, that the technical means employed meet the objectives proposed by the standards.

2.1. Safety

Since the publication of 1961 Edition of IAEA “Transport Regulations for the safe transport of radioactive material” [5], Argentina has adopted this document to regulate the transport of such materials. In 1994 it was made effective through the application of National Standard AR 10.16.1 “Transport of radioactive materials” which has been literally taken from the Spanish version of IAEA Regulations TS-R-1. In this field, the objective of ARN is intended to ensure protection and safety of people, property and the environment from the effects of ionizing radiation during the transport of RAM.

At present, the transport of RAM must be undertaken in accordance with the provisions set in Revision 2 of Standard AR 10.16.1 [6] which text concurs with that of the 2009 Edition of the IAEA Regulations TS-R-1 [7]. ARN is the competent authority for the application of this standard. From May 11, 2011, Revision 2 of the mentioned standard is in force in the country, with the following considerations:

a) There will be a transition period until December 31, 2011, during which the transport of RAM may be ruled optionally by either Revision 1 or Revision 2 of Standard AR 10.16.1. Revision 1 of the cited standard matches the text of the 1996 Edition (Revised) of IAEA TS-R-1 Regulations.
b) From January 1, 2012, Revision 1 of Standard AR 10.16.1 will no longer be valid.

There are also national and international regulations governing transport of dangerous goods by land, air and water, which related to radioactive materials, are consistent with IAEA’s TS-R-1 Regulations. Some of those regulations are:

1. The "Technical Instructions for the Safe Transport of Dangerous Goods by Air" [8] of the International Civil Aviation Organization (ICAO).
2. The "International Maritime Dangerous Goods Code" (IMDG Code) [9] of the International Maritime Organization (IMO).
3. The "Dangerous Goods Regulations" [10] of the International Air Transport Association (IATA) for transport of radioactive materials.

For the transport of RAM, Argentina pertinent Authorities follow IMO, ICAO and IATA regulations which adopted the 2009 Edition of IAEA's TS-R-1 Regulations in 2011.

2.2. Security

The objective of ARN related to physical protection is to prevent with a reasonable degree of certainty the theft, robbery, removal or dispersion as a malicious act, or unauthorized use of nuclear materials, or sabotage or intrusion of outsiders in nuclear facilities or during transport, which may generate accidents with severe radiological consequences due to their radioactive inventory. ARN has the responsibility of requiring the Operator a complete Physical Protection system for nuclear facilities and materials, and for radioactive facilities and sealed sources in accordance with the regulatory requirements set forth by ARN, as well as a Security Plan during transport of RAM.

Physical protection has become a matter of international interest and cooperation. In particular, the “Convention on Physical Protection of Nuclear Material”, regarding to international transport of these materials, was opened to signature on March 3, 1980, at IAEA’s headquarters in Vienna and the United Nations headquarters in New York; Argentina adopted this Convention under Act 23620 [11] and then ratified it, in the year 1988.

ARN carries out various activities related to the evaluation, monitoring and control of the design of the Physical Protection Systems and the Security Plans in the current regulatory framework set forth in Standard AR 10.13.1 "Standard for Physical Protection of Nuclear Materials and Installations", Revision 1 [12], and Standard AR 10.13.2 "Standard for Physical Protection of Sealed Sources", Revision 0 [13], which take into account the transport of RAM.

The Standard AR 10.13.1 is based on the IAEA INFCIR/225 “Nuclear security recommendations on physical protection of nuclear material and nuclear facilities”, being its first revision approved by CNEA in 1992. Currently the IAEA INFCIR/225 Revision 5 [14] is published.

In the case of the Standard AR 10.13.2, the radioactive sealed sources were categorized in accordance with the IAEA “Code of conduct on the safety and security of radioactive sources” [15] and the IAEA Safety Guide RS-G-1.9 “Categorization of radioactive sources” [16]. The IAEA Nuclear Security Series No. 9 “Security in the transport of radioactive material” [17] has been of great support in the implementation of some criteria of this standard.

3. Safety

3.1. Licensing and Control Tasks

Since the safety in transport of RAM has a strong dependence on the design of packages and materials, the licensing of such designs can be a complex process that requires many different skills.

Argentinean companies such as CNEA, INVAP S.E., DIOXITEK S.A., CONUAR S.A., POLYTEC RM S.R.L. and ASESORAMIENTO TECNOLOGICO S.R.L. have been involved in the design of special form radioactive material (SFRAM) and packages for over 30 years.

Transport of RAM is frequently carried out by the country. It can be mentioned that Argentina is the third largest producer of Co-60 following Canada and Russia. Currently, DIOXITEK S.A. manufactures sealed sources of Co-60 approved as SFRAM for domestic medical and industrial uses and also exports these sources and bulk Co-60 to ‘inter alia’ the United Kingdom, China, Uruguay, Chile, Bolivia and Venezuela.

It was the necessity of reducing the number of shipments of radioactive material approved under special arrangements and avoid as much as possible the dependence on foreign suppliers, what promoted the development of national package designs. Many times, this dependence made the transports more expensive or delayed them unnecessarily because it was impossible to have the required packaging in a timely manner for such transports. It is noted that as regulatory policy, ARN decided to reduce as much as possible the special arrangements.

In relation to licensing tasks for obtaining Approval Certificates according to Standard AR 10.16.1, ARN has expertise on verifying the compliance of such designs with the applicable requirements of mentioned standard, e.g., analysis, assessment and independent recalculation of shielding, criticality, radiological protection, materials. In particular, analysis and assessment related to dynamic impact and mechanical tests with scale specimens of package designs are evaluated by ARN external consultants.

Up to now, a total of 20 Approval Certificates of package and SFRAM designs have been issued; being 2 more designs in a licensing stage. Of that total, 10 belong to SFRAM models consisting of Co-60 and Ir-192 sealed sources for medical and industrial uses, and the other 12 certificates are package designs. These last are Type B(U) packages and Industrial, Type A and Type B(U) packages for transport of fissile material.

It is important to note that the first design certificates issued by the Argentinean Competent Authority were the RA/0027/S in September 1986 for a sealed source (SFRAM) for medical use and the RA/0033/B(U)F for transport of research reactor fresh fuel elements in February 1988.

It is also important to mention that ARN is in process of licensing a Type B(U)F package design for transport of fuel elements, fresh or spent with enriched Uranium. This design is being developed under the framework of IAEA Regional Technical Cooperation Projects, among Argentina, Brazil and Chile. This project has allowed qualification and experience in such relevant design. There have been significant advances in the design which would make possible its approval by ARN in 2013.

Another significant comment is that as part of the licensing process of Type B(U) package designs carried out by the Department for Transport (UK Competent Authority), ARN was invited to collaborate verifying mechanical tests conducted in Argentina. That was appreciated by ARN as it helped to exchange views between the two competent authorities on the preparation, conduct and outcome of such tests.

With the objective of controlling and monitoring the compliance with the applicable requirements of safety Standard AR 10.16.1, ARN performs inspections and regulatory audits, and verifies preventive and corrective actions taken by designers, consignors and other related users.

3.2. Quality Management System

Sure of having developed its regulatory experience with a high level of confidence, ARN decided to implement its Quality Management System (QMS) in 2005 [18].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 citizens and the transparency of ARN actions.

ARN has established a Quality Manual [19] based on standard ISO 9001:2008 “Quality management systems – Requirements” [20] and follows the IAEA recommendations of “PDRP-6 – Quality management of the nuclear regulatory body” [21].

The ARN process related to safety in the transport of RAM called "Protection against ionizing radiation in the transport of radioactive materials", TMR process, obtained ISO 9001 certification on May 20, 2008. It has received and maintained two certifications, the Certificate for Management Systems given by the Argentine Normalization and Certification Institute (in Spanish, Instituto Argentino de Normalización y Certificación, IRAM) and the International Certification Network (IQNet).

The application of TMR process takes place by implementing the TMR Quality Plan [22] that follows the guidelines of the Quality Manual of ARN, as well as procedures, work instructions and forms developed for the necessary arrangements. This allows the generation of documents and performance indicator values that provide annual and quarterly status report on the results and proposals for improving the process.

3.3. Training provided and participation in IAEA activities

Since about 40 years, ARN has provided training in safety during transport of RAM, conducting national and interregional courses as well as developing the appropriate training material in Spanish for such courses (lectures notes, practical works, slides, presentations).

ARN personnel have been involved as lecturers during specific training courses on transport of radioactive material in the framework of the IAEA Latin America Programme, ARCAL: in Peru, Costa Rica and Chile in 1990. Additionally, ARN was involved as advisor in assessing on national regulations and in training courses on that matter in Guatemala, Brazil, Panama, Bolivia, El Salvador, Colombia and Venezuela, as well as in courses developed in Argentina at technical and professional level addressed to consignors, designers and security forces.

In cooperation with the IAEA, ARN co-ordinated and gave interregional training courses on transport of radioactive material addressed to Latin America and The Caribbean countries, developed in Buenos Aires, Argentina: 1999 (3 weeks), and 2000 and 2008 (two weeks).

Argentina is member of the IAEA Transport Safety Standards Committee (TRANSSC), and also of the working group formerly known as Senior Advisory Group on Safe Transport of Radioactive Material (SAGSTRAM). Additionally, there are experts in transport of RAM who have participated in IAEA related matter meetings in IAEA Headquarters in Vienna. ARN experts have collaborated in the development of the IAEA Transport Regulations and others related support documents as well as their Spanish translations.

ARN Senior experts took part of the IAEA Transport Safety Appraisal Service (TranSAS) missions in United Kingdom (2002) and Panama (2003). Since 2006 ARN is member of the International Steering Committee on Denials of Shipments of Radioactive Material that is co-ordinated by IAEA.

3.4. Data Bases

The IMPO/EXPO Data Base is used for supporting and storage of relevant data of the endorsement by ARN of the Import and Export Application Form submitted by users. This Data Base records the user data (applicants), the probable date of importation or exportation, identification marks of approval certificates, amount and type of packages and radioactive contents involved in the importation or exportation.

The SHIP Data Base is used for recording the data related to Notice of RAM Transport forms submitted by users allowing, in this way, to have the orientative information about the quantity of land, air and water transports of these materials in Argentina. They are recorded the following data: consignor, consignee, origin, destination, carrier, radionuclides (physical form, activity, SFRAM or not), quantity, models and approval certificates of packages, transport index, route, starting and ending date and time, quantity of vehicles, and whether security measures are required.

4. Security

4.1. Tasks

It is noteworthy that since 1986 the first transports of Co-60 from Embalse Nuclear Power Plant (CNE) have been made under Gendarmería Nacional Argentina (GNA) custody, even before physical protection requirements were established by the National Regulatory Body and IAEA.

According to Standard AR 10.13.1, the protected nuclear materials (U-233, U-235, Pu-239, Pu-241) as well as the security measures that must be taken, are classified in three categories (I, II and III), depending on the amount of material in question. In the case of **Category I**, which is the most restrictive, the transport must be done with special care, i.e. the previous arrangement between consignor, consignee and carrier, specifying the time, place and procedures for transferring transport responsibility, detailing the mode of transport, routes to be used and reporting points in transit if necessary, and also under constant surveillance by escorts and under conditions that ensure communication at any time with the response personnel. One example can be mention, the transport of the spent fuel elements (high enriched Uranium) from the research reactors RA-3 (Ezeiza Atomic Center) and RA-6 (Bariloche Atomic Center) to USA, in the years 2000 and 2007 respectively, both conducted by Nac International Ltd, USA Department of Energy (DOE) contractor.

In the case of Standard AR 10.13.2, sealed sources as well as appropriate security measures, can also be classified in three categories (1, 2 and 3), depending on the activities of the sources to be transported. However, only the transport of **Category 1 Sealed Sources** must be carried out under security provisions according to ARN regulations, which shall be commensurate with the risks associated with each transport conditions, including if necessary the surveillance of each consignment and projections of the actions of the response personnel. Information on these projections must be of appropriate confidentiality. In the case of trans-boundary movements of this source category, it must ensure the continuity of security measures during transport and storage in transit and during the crossing of the border, complying also the requirements established by ARN for the importation or exportation of radioactive sources. For example, the transport of Co-60 teletherapy sealed sources approved as SFRAM from the production facility (Dioxitek S.A.) to the medical centres and points for international distribution must fulfil the above mentioned conditions.

Some of the measures to be taken for protected nuclear material and sealed sources are:

- Minimize the total time of the transport of material;
- Protect the material according to the category;
- Avoid regular routes;
- Confidentiality of information;
- Cross checks about "personnel reliability";
- Custody vehicles;
- Permanent communication;
- Satellite tracking.

As security procedures in general, consignors must submit to ARN the Security Plan for transport including satellite tracking systems, custody escort in their own vehicles, real time notification of departure and arrival, and any news that may occur during the journey, one responsible for the security designated by the consignor and a contact phone number. This person must have adequate capacity to receive information from the company providing satellite tracking, and to organize the initial response and notify the authorities and security forces about the news produced in transport. The consignors may contact the ARN security section in order to consult and agree the details of the requirements related to satellite tracking systems and custody escorts.

In the procedure previous to the transport of Category I nuclear material or Category 1 radioactive material, the company must submit to ARN a written security plan containing the details above mentioned with due anticipation and must also send the Notice of RAM Transport form, adding the corresponding security information.

In case of significant news or emergency, the intervention procedures should be applied in site for radiation protection purposes and the appropriate police authorities should be notified. In both cases the SIER (Sistema de Intervención de Emergencias Radiológicas de la ARN; in English, ARN Radiological Emergencies Intervention System) must be immediately informed. In addition to a direct intervention of SIER, when applicable, the company will also collaborate with the authorities doomed to solve the situation generated.

These procedures and security measures are used for both transports within Argentina, as well as transport related to imports and exports of sealed sources and nuclear material.

With the objective of controlling and monitoring the compliance with the applicable requirements of security Standards AR 10.13.1 and AR 10.13.2, ARN performs inspections and regulatory audits to consignors, carriers, and other related users.

4.2. Data Bases and Quality Management System

The Security section in ARN has implemented a Data Base which stores all information relevant to transport of RAM and the corresponding security measures: consignor, security, responsible, origin and destination of shipments, routes, type and amount of radionuclides, satellite tracking and custody company, are examples of the information stored.

ARN Security section acts as representative responsible for the IAEA Illicit Trafficking Database (ITDB). In addition, in the framework of Mercado Común del Sur (MERCOSUR; in English, South America Agreement of Partial Reach), this section acts as a contact point in the working group “Illicit trafficking in nuclear and/or radioactive material”.

Regarding to Quality Management System, the Security process contemplates some procedures related to inspections. Achieving ISO 9001 Certification is a medium-term objective in this section.

4.3. Training provided

At border crossings considered relevant, ARN has given specific training courses in transport security aimed to the personnel of the General Administration of Customs. This kind of courses is also given periodically to the different security forces.

In cooperation with the IAEA and the USA DOE - National Nuclear Security Administration (NNSA), ARN co-ordinated and gave a regional training course on Security of Radioactive Sources addressed to Latin America and The Caribbean countries, developed in Buenos Aires, Argentina, in 2006. This course included security in transport of RAM as a topic.

5. Safety and Security working together

Interfaces between Safety and Security sections are most of the times complementary. At the beginning it could be conflictive. However, during the last years both sections have integrated their tasks taking into account the different historical evolution they have experienced.

Currently, joint inspections covering both safety and security issues are performed. Sharing information, database, historical records, risks and threats are carried out in order to unify fulfilment for safety and security requirements.

The integration between both areas is also noticed in training tasks aimed to different sectors involved in the transport of RAM. Several courses covering safety and security issues are given.

The transport of large amounts of Co-60 from Embalse Nuclear Power Plant (CNE) to a sealed sources production facility (Dioxitek S.A.) is an example of a shipment that is routinely done in this country.

According to Standard AR 10.13.2, these sources are classified as Category 1 and according to Standard AR 10.16.1, a Type B(U) package must be used.

In this case, the consignor (Dioxitek S.A.) submits the Notice of RAM Transport form to the TMR section in ARN at least 48hs in advance to start the transport. Simultaneously, the consignor submits the Security Plan to the Security section in ARN. In this plan the consignor commits himself to inform ARN Security mobile phone of the departure and arrival of the conveyance as well as any news on the route of the shipment.

After checking the Notice of RAM Transport form, TMR section asks Security and Emergency sections for the consent to carry out the mentioned transport. In case of any of the sections involved does not provide such conformity, the transport must not start until the consignor complies with all pertinent requirements.

Before starting the shipment, the security responsible must notify the Security section in ARN of the inminent departure, and must do the same in case of any relevant news related to security. That section will be the responsible for monitoring the mentioned shipment. In case of emergency the security responsible must immediately contact Emergency ARN section, SIER, using the proper procedure.

6. Future plans

It can be taken as a vantage that both sections, safety and security, coexist in the same institution. ARN has successfully established a safety regulation system for the transport of RAM, whereas transport security has been developed and integrated into the national regulatory framework.

It is important to conduct practical activities in order to promote safety and security in a synergetic manner. According with this point of view, the following actions should be taken in order to achieve further improvement:

- It is important to continue working on the harmonization and integration of the safety and security areas in transport in order to develop a strong Safety and Security Culture, taking into account the different historical evolution that both areas have experienced. It is intended to work on safety and security in an efficiently and effectively way, without duplication or conflict.
- Joint inspections covering both safety and security issues will be continued, as well as sharing information, database, historical records, risks and threats; these would promote the unified fulfillment for safety and security requirements.
- Further training and awareness of all sections involved in Transport should be strengthened. In particular, ARN will continue giving specific courses covering both topics and directed to security forces and other similar audiences.
- It is expected to achieve ISO 9001 Certification in the section of Security in a medium-term; this would be in line with the certification of the different processes of ARN.

- In the framework of IAEA, ARN will continue working in order to solve the problems which interfere with the transport of RAM, and could result in delays or denials of shipments.
- Taking into account Argentine expertise on transport safety and security as well as its experience in the preparation and/or modification of IAEA related documents, try to share it with Latin America and The Caribbean countries promoting regional events.

7. Conclusions

Considering the experience acquired since the beginning of nuclear activity in Argentina in 1950 and essentially the proceedings of ARN during the last decade, as conclusions of this work it can be mentioned that it is important:

- To participate in the elaboration or modification of IAEA Transport documents, and to implement and improve the Argentine ARN standards as well as maintain them updated as far as possible in line with such documents.
- It would be encouraging that a harmonization of documents and especially the vocabulary used in safety and security was prompted by the IAEA, in order to facilitate the interpretation and application of standards and guidelines that are published.
- To improve communication and joint co-ordination between safety and security sections in ARN to assure appropriate accomplishment of their objectives.
- In the framework of implementation of a quality management system according to ISO 9001 standard, to keep the continuous improvement tasks related to Safety and Security in Transport.
- To promote, through TRANSSC and Security meetings, the activities related to the harmonization in the implementation of regulatory requirements among competent authorities, regulatory bodies and international modal organizations.

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REFERENCES

[1] Decreto N° 10936. Creación de la CNEA, Argentina, 1950.
[2] Decreto N° 1540. Creación del ENREN, Argentina, 1994.
[3] Ley N° 24804. Ley Nacional de la Actividad Nuclear, Argentina, 1997.
[4] Decreto N° 1390. Reglamentario de la Ley Nacional de la Actividad Nuclear, Argentina, 1998.
[5] INTERNATIONAL ATOMIC ENERGY AGENCY. Regulations for the safe transport of radioactive material, Safety Standards No.6, 1961 Edition. IAEA, Vienna, 1961.
[6] AUTORIDAD REGULATORIA NUCLEAR, Transporte de materiales radiactivos, AR 10.16.1, Revisión 2. ARN, Buenos Aires, Argentina, 2009.
[7] INTERNATIONAL ATOMIC ENERGY AGENCY. Regulations for the safe transport of radioactive material, Safety Standards Series No.TS-R-1, 2009 Edition. IAEA, Vienna, 2009.
[8] INTERNATIONAL CIVIL AVIATION ORGANIZATION. Technical instructions for the safe transport of dangerous goods by air. ICAO. 2009-2010 Edition, Montreal, 2009.
[9] INTERNATIONAL MARITIME ORGANIZATION. International maritime dangerous goods code (IMDG Code). IMO. Amendment 35-10, 2010 Edition, London, 2010.
[10] INTERNATIONAL AIR TRANSPORT ASSOCIATION. Dangerous goods regulation. IATA, 2011. 52nd Edition, Montreal and Geneva, 2010.
[11] Ley N° 23620. Convención sobre protección física de los materiales nucleares, Argentina, 1989.
[12] AUTORIDAD REGULATORIA NUCLEAR, Norma de protección física de materiales e instalaciones nucleares, AR 10.13.1, Revisión 1. ARN, Buenos Aires, Argentina, 2002.
[13] AUTORIDAD REGULATORIA NUCLEAR, Norma de seguridad física de fuentes selladas, AR 10.13.2, Revisión 0. ARN, Buenos Aires, Argentina, 2007.
[14] INTERNATIONAL ATOMIC ENERGY AGENCY. Nuclear security recommendations on physical protection of nuclear material and nuclear facilities, INFCIRC/225/Rev.5. IAEA, Vienna, 2011.
[15] INTERNATIONAL ATOMIC ENERGY AGENCY. Code of conduct on the safety and security of radioactive sources. IAEA, Vienna, 2004.
[16] INTERNATIONAL ATOMIC ENERGY AGENCY. Categorization of radioactive sources, RS-G-1.9. IAEA, Vienna, 2005.
[17] INTERNATIONAL ATOMIC ENERGY AGENCY. Security in the transport of radioactive material, Nuclear Security Series No. 9. IAEA, Vienna, 2008.
[18] Management system for regulating transport of radioactive material, J. López Vietri, N. Capadona and L. Barenghi. IRPA 12, Buenos Aires, Argentina, 2008.
[19] AUTORIDAD REGULATORIA NUCLEAR, Manual de Calidad ARN, MC-ARN, Revisión 03, Argentina, Buenos Aires, 2010.
[20] INTERNATIONAL ORGANIZATION FOR STANDARDIZATION, Quality management systems – Requirements, ISO 9001:2008, ISO, Geneva, 2008.
[21] INTERNATIONAL ATOMIC ENERGY AGENGY, Quality management of the nuclear regulatory body, PDRP-6, IAEA, Vienna, 2001.
[22] AUTORIDAD REGULATORIA NUCLEAR, Plan de Calidad TMR, PC-TMR, Revisión 3, Argentina, Buenos Aires, 2010.

HLA-G Modulates the Radiosensitivity of Human Neoplastic Cells

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Favier, B. and Carosella, E.D.

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HLA-G MODULATES THE RADIOSENSITIVITY OF HUMAN NEOPLASTIC CELLS

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INTRODUCTION

Tumor cells show a very broad range of radiosensitivities. The differential radiosensitivity may depend on many factors, being the efficiency to recognize and/or repair the DNA lesion, and the cell cycle control mechanisms, the most important (Jeggo and Lavin, 2009; Kumala et al., 2003).

Human leukocyte antigen-G (HLA-G) is a non-classical HLA class I molecule involved in fetus protection form the maternal immune system, transplant tolerance, and viral and tumoral immune escape (Carosella et al., 2008).

It has been determined that gamma radiation modulates HLA-G expression at the plasma membrane of human melanoma cells. However, its role in tumoral radiosensitivity has not been demonstrated yet.

The objective of this work was to determine if the radiosensitivity of human neoplastic cell lines cultured in vitro was mediated by HLA-G expression.

MATERIALS AND METHODS

Cell culture and γ-Irradiation: We used two different cell lines: M8 human melanoma cells and K562 human leukaemia cells. For both cell lines we have the HLA-G1 negative (G-) and the HLA-G1 positive (G+) variants. The cells were irradiated with doses of 2 and 5 Gy at 0.48 Gy/minute with ^{60}Co γ-rays generated by Gammacel 220 at room temperature.

Survival assays: cell survival was evaluated by clonogenic assay for the M8 cells and by direct counting in Neubauer chamber for the K562 cells

Detection of apoptotic cells: Apoptosis levels were determined by flow cytometry using the Annexin V-FITC/Propidium iodide kit. Results were expressed as the ratio between the percentage of apoptotic cells in the irradiated condition and the percentage of apoptotic cells under control conditions.

Cell cycle analysis: Samples were collected at different times after irradiation and fixed in ethanol 70% (v/v). The cells were then resuspended in PBS 1X containing 100 μg/ml RNAse and 40 μg/ml propidium iodide and assessed for cell-cycle distribution by flow cytometry.

Evaluation of HLA-G1 surface expression: Surface HLA-G1 expression was analyzed by flow cytometry after 1 and 15 days of irradiation. The cells were labeled with anti-HLA-G (MEM-G/9) monoclonal antibody in combination with goat anti-mouse-RPE. HLA-G1 surface values were expressed as specific fluorescence index (SFI; mean fluorescence of specific antibody/mean fluorescence of isotype antibody) and the ratio (SFI irradiated cells/SFI control cells) was used to express the level of HLA-G1 at the surface of irradiated cells.

Analysis of sHLA-G1: The presence of sHLA-G1 in the cell culture medium of control and irradiated cells was analyzed using a specific ELISA kit.

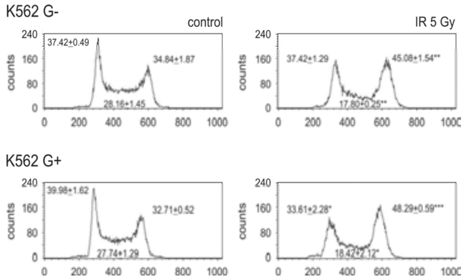
Table 1. Cell survival (percentage) of HLA-G expressing cells (G+) with respect to HLA-G negative cells (G-) after irradiation with the indicated doses of g radiation. *p < 0,001 (Student's t-test) with respect to G- (100%).

Cell line	% of cell survival of G+ with respect to G- cells	
	2 Gy	5 Gy
M8 G+	49,53 ± 22,83 *	49,06 ± 15,91 *
K562 G+	--	62.50 ± 11.48 *

Table 2. Cell cycle distribution of M8 cells after the indicated times of irradiation with 5 Gy. * p< 0.05; ** p< 0.005; *** p< 0.001 (ANOVA) with respect to control values. # p< 0.05 (ANOVA) with respect to M8 G-.

	G0/G1	S	G2/M
M8 G-			
control	58.94 ± 3.9	16.53 ± 3.2	23.96 ± 1.4
3 h	52.63 ± 9.8	20.12 ± 4.0	27.37 ± 5.5
24 h	5.68 ± 0.2 *	2.97 ± 1.6 ***	91.4 ± 1.7 *
48 h	10.37 ± 1.7 *	3.61 ± 0.9 ***	86.07 ± 2.5 *
196 h	44.31 ± 5.3 *	25.48 ± 4.4	28.78 ± 0.4 **
M8 G+			
control	58.79 ± 4.7	15.60 ± 3.8	25.57 ± 0.7
3 h	57.67 ± 12.9	18.92 ± 4.9	23.41 ± 6.6
24 h	3.55 ± 0.4 *	1.84 ± 1.1 ***	94.65 ± 1.0 * #
48 h	11.73 ± 0.6 *	5.13 ± 0.6 **	83.16 ± 1.2 *
196 h	47.12 ± 1.9	22.02 ± 0.6	30.32 ± 1.4 **

Figure 1. Cell-cycle distribution of K562 G- and G+ cells exposed to 5 Gy and analyzed after 24 h post-irradiation. One plot representative of three experiments is shown. *p< 0.005; **p< 0.001; ***p< 0.0001 (ANOVA) with respect to control values.



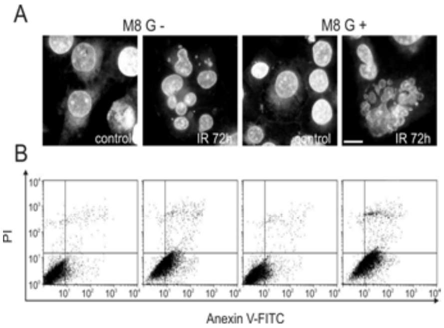


Figure 2. A) Hoechst 33342 labelling of control and irradiated M8 G- and G+ cells, analyzed after 72 h. Scale bar: 25 μm and 12,5 μm in control and irradiated cells respectively. B) Annexin V-FITC/PI flow cytometry dot-blots corresponding to M8 G- and G+ cells exposed to the same experimental conditions described in (A).

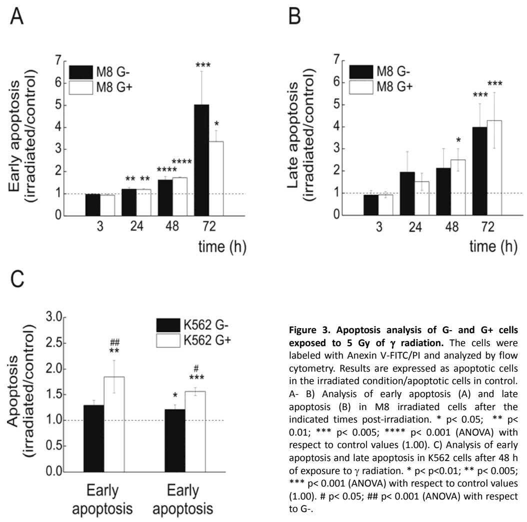


Figure 3. Apoptosis analysis of G- and G+ cells exposed to 5 Gy of γ radiation. The cells were labeled with Anexin V-FITC/PI and analyzed by flow cytometry. Results are expressed as apoptotic cells in the irradiated condition/apoptotic cells in control. A- B) Analysis of early apoptosis (A) and late apoptosis (B) in M8 irradiated cells after the indicated times post-irradiation. * p< 0.05; ** p< 0.01; *** p< 0.005; **** p< 0.001 (ANOVA) with respect to control values (1.00). C) Analysis of early apoptosis and late apoptosis in K562 cells after 48 h of exposure to γ radiation. * p< p<0.01; ** p< 0.005; *** p< 0.001 (ANOVA) with respect to control values (1.00). # p< 0.05; ## p< 0.001 (ANOVA) with respect to G-.

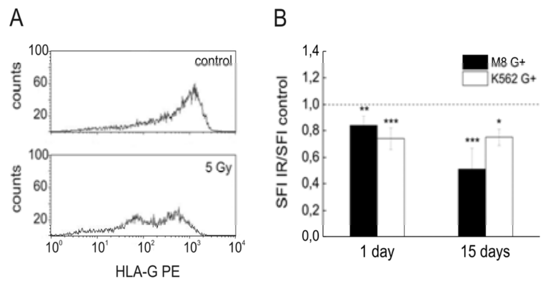


Figure 4. Flow cytometry evaluationell surface HLA-G in control and irradiated cells. A) Histograms of HLA-G surface expression in M8 G+ cells under control and irradiated (5 Gy) conditions, 15 days after irradiation. One histogram representative of three experiments is shown. B) Cell surface HLA-G quantification in M8 G+ and K562 G+ cells after 1 and 15 days of irradiation with 5 Gy. Results are expressed as SFI IR/SFI control. * p < 0.05, ** p < 0.01, and *** p < 0.0001 (Student's t-test) with respect to control values (1.00). **of c**

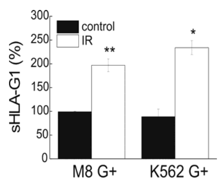


Figure 5. Concentration of sHLA-G1 in the culture medium of control and irradiated M8 G+ and K562 G+ cells after 24 h of irradiation with 5 Gy. Results are expressed as the percentage of sHLA-G1 in control and irradiated cells * p< 0.05 and **p< 0.01 (Student's t-test) with respect to controls (assigned a value of 100%).

CONCLUSIONS

• Cell survival after irradiation was significantly reduced in HLA-G1 expressing cells with respect to G- cells.

• Irradiation caused arrest of both cell lines in the G2/M phase of the cell cycle.

• The percentage of arrest after 24 h of irradiation in M8 G+ cells was a statistically higher than in M8 G- cells.

• For the K562 cell line, the magnitude of the arrest was lower than for M8 cells and without statistically significant differences between K562 G- and K562 G+ cells.

• Both cell lines underwent apoptosis after irradiation. However the level of apoptosis between M8 G- and M8 G+ was not different, whereas for K562 cells the G+ cell line reached higher apoptosis values than the K562 G-.

• These results could indicate that both the M8 G+ and the K562 G+ are indeed more radiosensitive than their corresponding G- cells, although the mechanism through which HLA-G1 exerts this effect in each cell line appears to be different.

•Gamma irradiation also caused the decrease of HLA-G1 levels at the cell surface of G+ cells with the concomitant increase of sHLA-G1 in the culture medium.

The results presented in this work strongly demonstrated that HLA-G1 can modulate the radiosensitivity of human tumoral cells culture in vitro. However, further studies will be necessary to asses the precise mechanism by which HLA-G1 exerts this regulation of radiation sensitivity.

References

✓Jeggo P, Lavin MF. Cellular radiosensitivity: how much better do we understand it? Int J Radiat Biol. 2009. 85(12):1061-1081.

✓Kumala S, Niemiec P, Widel M, Hancock R, Rzeszowska-Wolny J. Apoptosis and clonogenic survival in three tumour cell lines exposed to gamma rays or chemical genotoxic agents. Cell Mol Biol Lett. 2003. 8(3):655-65.

✓Carosella ED, Moreau P, Lemaoult J, Rouas-Freiss N. HLA-G: from biology to clinical benefits. Trends Immunol. 2008. 29(3):125-132. Review.

Inhibition of Cellular Growth of Melanoma Cells Line by Heterogeneous Beta Cronic Irradiation at Very Low-Dose-Rate

Michelin, S.; Gallegos, C.; Dubner, D.L.;
Favier, B. and Carosella, E.D.

Presentado en: Third International MELODI Workshop.
Roma, Italia, 2 al 4 de noviembre de 2011

INHIBITION OF CELLULAR GROWTH OF MELANOMA CELLS LINE BY HETEROGENEOUS BETA CRONIC IRRADIATION AT VERY LOW-DOSE-RATE.

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INTRODUCTION

Radioisotopes that decay via beta emission are widely used in science and medicine. The main advantage of beta-emitters is the relatively long path length in biological tissue (in the mm range).

The objective of this work was to determine the inhibition of cellular growth of melanoma cells (melanoma cells are one of the most radioresistant tumor cells) by beta irradiation at very low dose rate using a simple and economic device. This irradiation system represents a situation similar to radiodiagnostic, radioimmunotheraphy and brachytherapy because there is a continuous emission of exponentially decreasing low-dose-rate irradiation with heterogeneous dose deposition. It permitted us to study different dose rates changing only the activity of beta emitter. We compare it with the high dose rate gamma irradiation.

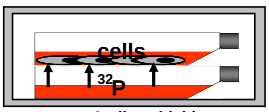
MATERIALS AND METHODS

Cell culture: Human melanoma cell lines, FON and M8 were used. The cells were seeded 1 day before the start of the irradiation on 25 cm^2 tissue culture flasks and the final cell number was determined at the end of irradiation period (4 to 6 days).

Gamma irradiation: Both cells line were irradiated with 2 Gy at 30.000 mGy/h with Gammacell 220 at room temperature and 6 days post irradiation the number of viable cells was determined.

Beta irradiation system: The system is composed of two identical tissue culture flasks superposed. The bottom flask (irradiation flask) was loaded by adding 150 µCi of ^{32}P orthophosphate solution in 3 ml of water (Figure 1). The absorbed dose in the upper culture flask and the isodose curves were calculated applying MCNPX 2.5f Monte Carlo code coupled to photon and electron cross sections from ENDF/B-VI library and validated by Gafchromic EBT2 film dosimetry. The irradiation and cell culture system was kept in incubator at 37 C and with 5% CO_2 until the total dose was delivered. The final dose was 2 Gy and the initial dose rate was 12-15 mGy/h.

Figure 1. Schematic conditions of irradiation



Cell cycle analysis of M8 cells: Samples were collected after 24, 48 and 72 h of beta irradiation and fixed in ethanol 70% (v/v). The cells were resuspended in PBS 1X containing 100 µg/ml RNAse and 40 µg/ml propidium iodide and assessed for cell cycle distribution by flow cytometry.

RESULTS

Dose rate distribution and final dose: Figure. 2 show the initial dose rate distribution in the flask. The initial average dose rate was 0,122 mGy/h/µCi. The final dose ranged from 100% in the center of the flask, to 67% at the external limit. The value of dose and dose rate shown represent the average value calculated by the Monte Carlo code. The final dose was obtained after 6 days or irradiation.

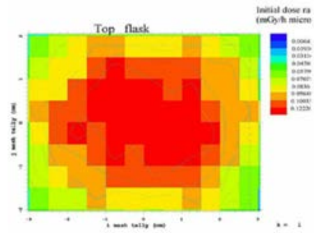


Figure 2. Initial dose rate distribution in the culture flasks.

Cell survival: the cell survival was determined at the end of irradiation period for beta irradiation and at the same period after acute gamma irradiation. (Table 1).

Table 1. Cell survival of FON and M8 melanoma cells after beta and gamma irradiation.

Cell line	Final Dose (Gy)	Dose rate (mGy/h)	Diminution respect to the control	Irradiation source
FON	2	12-15	28,61 ± 7.6	beta
M8	2	12-15	40,75 ± 5,8	beta
FON	2	30.000	28,33 ± 9,5	gamma
M8	2	30.000	45,22 ± 7,0	gamma

Cell cycle analysis: cell cycle distribution of M8 cells was analysed by flow cytometry as discribed in Materials and Methods after 24, 48 and 72 h of beta irradiation. Results are shown in Table 2.

Table 2. Cell cycle distribution of M8 cells after the indicated times of beta irradiation. * $p< 0,05$; ** $p< 0.005$ (Student t test) with respect to control values.

	Total dose (Gy)	G0/G1	S	G2/M
control 24 h	--	48,60 ± 1,54	21,51 ± 0,63	30,10 ± 1,88
IR 24 h	0.40	46,93 ± 1,55	20,73 ± 0,91	32,55 ± 0,74
control 48 h	--	46,75 ± 2,60	24,95 ± 1,69	28,52 ± 0,98
IR 48 h	0.78	44,39 ± 1,51	20,79 ± 2,55	35,02 ± 1,15 **
control 72 h	--	52,79 ± 2,18	19,41 ± 0,36	27,98 ± 1,91
IR 72 h	1.14	44,92 ± 0,93 **	20,22 ± 0,15 *	35,06 ± 0,84 **

CONCLUSIONS

We determine the effectiveness in producing significant cell death of two melanoma cell lines at very low dose rate beta irradiation (12-15 mGy/h) and we compare this effect with high dose rate gamma irradiation (30.000 mGy/h). This last dose rate is in the range of the dose rate used in radiotherapy. Gamma irradiation was employed because the relative biological efficiency is similar to beta irradiation.

For FON and M8 cell lines, cell survival diminution respect to controls was significant and independent of the dose rate.

Beta irradiation caused the progressive arrest of M8 cells in the G2/M phase of the cell cycle (Table 2).

Studies are in progress to determine if the mechanisms that influence cell killing for low and high dose rate are different and if the bystander effect is involved in this response at very low dose rate.

This irradiation system is simple, economic, and it is possible to obtain different dose rate, changing the ^{32}P activity.

Our results would be a useful contribution for the optimization of radiotherapy protocols.

Environmental Impact in Argentina of the Nuclear Tests

Quintana, E.

Presentado en: Conferencia de Ciencia y Tecnología.
Viena, Austria, 8 al 10 de junio de 2011

Nuclear Regulatory Authority

Argentina

www.arn.gob.ar

ENVIRONMENTAL IMPACT IN ARGENTINA OF THE NUCLEAR TESTS

Quintana Eduardo

Nuclear Regulatory Authority (Autoridad Regulatoria Nuclear, ARN) - Argentina

INTRODUCTION

During the period 1945 - 1980 more than 400 atmospheric nuclear weapon tests were performed. Before that, the fallout was almost exclusively originated in the stratosphere. Radioactive material generated in a nuclear explosion can be divided in 3 fractions. Mayor particles, that are deposited close to explosion site. Small particles that remain longer in the troposphere and are dispersed and transported by the wind. These particles are gradually deposited, almost along the same longitude. The last and more important fraction reaches the stratosphere were they spread, falling slowly on the earth. This fallout is composed of hundred of radionuclides but 4 of them (14C, 137Cs, 90Sr and 3H) contributed mainly to the exposition of humans. A statistical procedure was used to analyze the temporary variation of the 90Sr and 137Cs concentrations. The obtained results allowed to assess the environmental impact of the radioactive fallout in Argentina due to past nuclear explosions.

OBJECTIVE

- The objective of this research is to increase the knowledge of the contamination caused by the atmospheric nuclear weapon tests in the South Pacific carried out in the past.

ENVIRONMENTAL MONITORING

SAMPLING:

- Samples of deposition and fresh milk have been taken in the city of Buenos Aires and the surroundings since 1960.
- Due to the fact that the activity concentration decreased with time, the period of sampling changed accordingly.
- ***Milk:***
- Daily / Weekly / Monthly
- ***Deposit:***
- Weekly / Monthly / Quarter

MEASUREMENT OF 137Cs:

- Initially NaI (Tl), afterward Ge(Li) were used and presently HPGe detectors.
- Different geometries were used according to the radioactivity found.
- ***Milk:***
- 1 Liter (directly)
- 60 gr cylinder of milk ash.
- Disc, obtained by radiochemistry precipitation.
- ***Deposit:***
- 30 cm3, previous evaporation.
- Disc, obtained by radiochemistry precipitation.

Hiperpure Germanium Detectors

Liquid Scintillator Equipment

MEASUREMENT OF 90Sr:

- Initially Geiger-Muller (thin window), afterward plastic detectors were used and presently liquid scintillator equipment.
- Radiochemistry separation was used from the beginning for milk and deposit.

METHODOLOGY:

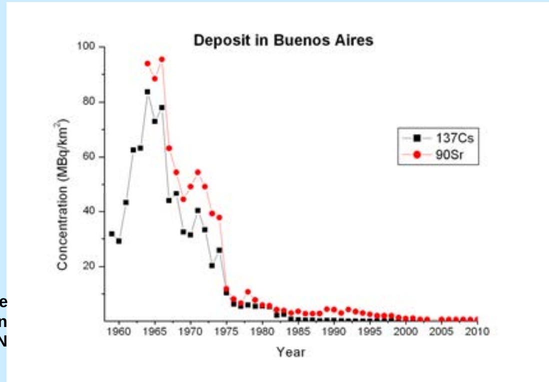
- The distribution of data from the sixties till today were determined using goodness of fit (Kolmogorov-Smirnov). The data analyzed annually fitted a log-normal distribution.
- Normalized data of the mentioned samples and radionuclides were compared using student test (two-tailed).

RESULTS :

- From the mentioned studies in Buenos Aires, two sets of data are clearly distinguished, one during the nuclear weapon tests era and another one after it.
- Radioactive fallout from the stratosphere and troposphere generated important peaks in 1964, 1965 and 1966.
- Secondary peaks of tropospheric fallout came from the South Pacific tests in 1970, 1971 and 1972.

- ***90Sr:***
- The maximum concentrations of 90Sr in deposit and milk were registered in 1964. Concentrations of 90Sr activity in deposit and milk were 83.6 MBq/km2 and 240.8 mBq/liter.
- Today, the corresponding concentrations are less than 0,02 MBq/km2 and 10 mBq/liter respectively.

- ***137Cs:***
- The maximum concentrations of 137Cs in deposit and milk were measured in 1966. Concentrations in deposit and milk were 95 MBq/km2 and 944 mBq/liter.
- Today, these values are less than 0.60 MBq/km2 and 4.7 mBq/liter respectively.



<u>Data Source:</u> For the development of the work, the data were extracted from the following sources: Technical Notes of Comisión Nacional de Energía Atómica, ARN Annual Reports and ARN Internal Reports (non published).

CONCLUSION

The results show that the environmental concentration has been decreasing during the past decades although at a slower rate in recent years. This effect is only observed because of the long term sampling carried out. Nowadays, the radioactivity concentrations are very low and close to reaching stability and the observed oscillations are due to re-suspension phenomenon.

For more information submit an e-mail to ctbtlab@arn.gob.ar

Calibración de equipos de laboratorios y su verificación intermedia

Remedi, J.O.

Presentado en: XXI FORO UNILAB.
La Plata, Argentina, 10 al 11 de noviembre de 2011

CALIBRACIÓN DE EQUIPOS DE LABORATORIOS
Y SU VERIFICACIÓN INTERMEDIA

Remedi, J.O.[1]

Autoridad Regulatoria Nuclear
Argentina

1. RESUMEN

Cuando un laboratorio quiere probar que posee competencia técnica para la realización de ensayos o calibraciones deberá demostrar que ha dado cumplimiento a ciertos requerimientos que establecen, entre otros, la obligatoriedad de: calibrar o verificar un equipo antes de ponerlo en servicio a efectos de asegurar que responde a las especificaciones establecidas del laboratorio; llevar registros de los equipos que evidencien las verificaciones de la conformidad del equipo con la especificación; realizar comprobaciones intermedias para mantener la confianza en el estado de calibración de los equipos; asegurar que se comprueban el funcionamiento y el estado de calibración del equipo cuando el equipo quede fuera del control directo del laboratorio, antes de ser reintegrado al servicio; establecer un programa y un procedimiento para la calibración de sus equipos; demostrar como determina los periodos de calibración de sus equipos así como las evidencias de que las verificaciones intermedias son adecuadas a los periodos de calibración. Ahora bien, se observa cierta confusión en cuanto al significado de los términos "calibración" y "verificación" de un equipo. El presente trabajo, analiza la documentación aplicable y postula que las diferencias se generan en parte por las traducciones y también por la caracterización de los conceptos según sea su uso, esto es, si se trata de metrología legal o de evaluación de la conformidad. Por ello este trabajo se propone caracterizar ambos conceptos, acercar fundamentos para diferenciarlos, esbozar estrategias apropiadas para las actividades de calibración y verificación que aseguren el cumplimiento de los requisitos normativos aplicables.

Palabras clave: Calibración, verificación, patrones, trazabilidad, competencia.

2. INTRODUCCIÓN

Toda intención de realizar una medición en un laboratorio supone tomar decisiones en cuanto a los siguientes temas: elegir el equipo más adecuado para ese fin, decidir la mejor estrategia de calibración y posterior verificación, optar por los patrones más apropiados que provee el mercado y finalmente, resolver la mejor manera de dejar documentado el control sistemático que se ejerce sobre el equipo y el correcto uso que del mismo se ha hecho. Si se ha prestado atención a todo ello, el laboratorio podrá garantizar que la medición realizada satisface tres aspectos fundamentales: cumplir con las leyes y regulaciones en la materia; dar cuenta de la calidad de las mediciones al cliente y asegurar que las mediciones llevadas a cabo son trazables a patrones reconocidos. Estas exigencias son ineludibles para cualquier laboratorio de calibración y ensayo que aspire a ser acreditado conforme a la Norma IRAM 301:2005 (en adelante "la Norma") vigente en la actualidad. Ahora bien, de entre todas aquellas decisiones, el presente trabajo sólo se ocupará de las relativas a la "calibración" y a la "verificación", vocablos estos que suelen traer aparejada cierta confusión entre los agentes involucrados de un laboratorio, además de controversias con los evaluadores, proveedores y clientes. Así, tomando como base los puntos de la Norma y los Criterios del Evaluación del Organismo Argentino de Acreditación (OAA), que hablan de estos conceptos, se tratará de hacer un poco de luz sobre ellos, lo que permitirá esbozar algunos modelos para el mejor control y mantenimiento de los equipos de un laboratorio.

3. OBJETIVOS

Caracterizar las diferencias entre **calibración** y **verificación**.

Definir algunos criterios para determinar la periodicidad con que se realizarán **verificaciones** a los equipos del laboratorio.

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Y, en última instancia, esbozar algunas estrategias para demostrar que en estas dos materias, se cumplen con los requisitos de la Norma y de los Criterios del OAA (los Criterios).

4. METODOLOGÍA

Primero se verá en detalle, qué dicen al respecto, la Norma y los Criterios del OAA, analizando la consistencia terminológica en la redacción de estos documentos respecto de ambos vocablos. Posteriormente, se esbozarán algunas estrategias de calibración y verificación intermedia para un equipo y un período determinado. Finalmente se determinará si con esas estrategias se satisfacen los requisitos de la Norma y los Criterios del OAA.

5. ANÁLISIS

5.1 Análisis de Antecedentes

Comencemos por el análisis de los puntos relativos a la calibración y verificación de equipos que presenta la Norma.

La Norma contiene al menos seis puntos relativos a estos conceptos, a saber:

> "*5.5.2 … Antes de poner en servicio un equipo se debe* ***calibrar o verificar*** *con el fin de asegurar que responde a las exigencias especificadas del laboratorio y cumple las especificaciones normalizadas pertinentes. El equipo debe ser* ***verificado o calibrado*** *antes de su uso."*

Es decir, se pretende que el operador, inicie con plena confianza las operaciones de medición con el equipo asignado al efecto.

> "*5.5.5 Se deben establecer registros de cada equipamiento y su software que sean importantes para la realización de los ensayos o las* ***calibraciones****. Los registros deben incluir por lo menos lo siguiente:*
> …
> *c) las verificaciones de la conformidad del equipo con la especificación (ver 5.5.2)"* …

Es decir, el laboratorio debe contar con elementos objetivos, como es el caso de registros completos, para demostrar que los equipos usados para la medición satisfacen las especificaciones requeridas por la Norma o bien por el uso previsto.

> "*5.5.9 Cuando, por cualquier razón, el equipo quede fuera del control directo del laboratorio, éste debe asegurarse que se comprueban el funcionamiento y el estado de* ***calibración*** *del equipo y de que son satisfactorios, antes de que el equipo sea reintegrado al servicio".*

Esto es toda una apelación, que recuerda al punto 5.5.2, con la intención de asegurar que ninguna medición se realizará, si no se han obtenido las pruebas suficientes para la confiabilidad en la operación del equipo. Veamos además qué dicen los Criterios del OAA en este sentido:

> *C 5.6.2.1 Los certificados de calibración externa deberán haber sido emitidos por laboratorios de calibración acreditados por el OAA o por cualquier organismo de acreditación firmante de acuerdos de Reconocimiento Multilateral de ILAC (MLA) o por un Instituto Nacional de Metrología que participe satisfactoriamente en las intercomparaciones reconocidas por el BIPM.*

Este requerimiento, busca que el laboratorio tome ciertas precauciones cuando se contrata un laboratorio externo para que la calibración de equipos cumpla con la Norma y los Criterios. Continuamos,

> "*5.5.10 Cuando se necesiten* ***comprobaciones intermedias*** *para mantener la confianza en el estado de* ***calibración*** *de los equipos, éstas se deben efectuar según un procedimiento definido."*

Obsérvese que aquí ya no estamos hablando de calibración, sino de verificación. Pero lamentablemente la Norma ahora introduce la palabra "comprobación" en lugar de verificación. Y las actividades de "comprobación" que se sugieren, están referidas al control del equipo. No obstante, lo que se busca acá es que el laboratorio pueda demostrar que la calidad de las mediciones se mantiene desde la última calibración y además, que así puede continuar hasta la próxima calibración.

> "*5.6 Trazabilidad de las mediciones.*
> *5.6.1 … El laboratorio debe establecer un programa y un procedimiento para la* ***calibración*** *de sus equipos."*
>
> *NOTA: Es conveniente que dicho programa incluya un sistema para seleccionar, utilizar,* ***calibrar, verificar****, controlar y mantener los patrones de medición, los materiales de referencia utilizados como patrones de medición, y los equipos de ensayo y de medición utilizados para realizar los ensayos y las* ***calibraciones****"*

Aquí es donde empieza a tratarse el tema de la trazabilidad y se pretende que el responsable del laboratorio haya definido los periodos de calibración de sus equipos, los procedimientos con ese fin y demuestre que las verificaciones intermedias que realiza, conforme a un programa, son las adecuadas para los períodos de calibración establecidos. Luego de este repaso por ambos conceptos, tratados en la Norma y los Criterios del OAA, veamos ahora qué podemos decir de los conceptos que nos ocupan: **calibración y verificación**.

5. 2 Calibración

El Vocabulario Internacional de Metrología vigente, expresa una definición del primer concepto, como sigue:

> *"2.39* ***Calibración****: Operación que bajo condiciones especificadas establece* ***en una primera etapa*** *una relación entre los valores y sus incertidumbres de medida asociadas obtenidas a partir de las indicaciones de los patrones de medida y las correspondientes incertidumbres;* ***y en una segunda etapa****, utiliza esta información para establecer una relación que permita obtener un resultado de medida a partir de una indicación*."

Además expresa:

> *NOTA 1: Una* ***calibración*** *puede expresarse mediante una declaración, una función de calibración, un diagrama de calibración, una curva de calibración o una tabla de calibración. En algunos casos, puede consistir en una corrección aditiva o multiplicativa de la indicación con su incertidumbre correspondiente.*
>
> *Nota 2: Conviene no confundir la* ***calibración*** *con el ajuste de un sistema de medida, a menudo llamado incorrectamente "autocalibración",* ***ni con una verificación de la calibración****.*
>
> *Nota 3: Frecuentemente se interpreta que únicamente la primera etapa de esta definición corresponde a la* ***calibración****.*

Tomemos el caso de una balanza analítica. Es usual que un laboratorio confíe en los laboratorios del INTI o en un laboratorio acreditado, como proveedor del servicio de calibración. Su tarea será la de controlar el instrumento con patrones trazables a patrones internacionales o nacionales establecidos, con procedimientos estandarizados y validados. Si fuera necesario, realizará los ajustes necesarios al instrumento para dejarlo en las condiciones de funcionamiento apropiadas a los requerimientos establecidos o sugerirá dejarlo fuera de servicio, según sea el caso. El servicio contratado estimará el valor convencionalmente verdadero de una medida materializada y los errores de indicación del equipo entre otras características metrológicas, tales como sensibilidad, fidelidad, control del rango de uso, movilidad, error de excentricidad, histéresis, etc. Como producto de esa actividad, el órgano calibrador emite un Certificado de Calibración con tablas, gráficos, la determinación del error máximo permitido (o tolerado) - que veremos más adelante – y todos los detalles del trabajo realizado. Dicho certificado pasará a formar parte de la dotación de documentos del laboratorio, como evidencias objetivas para demostrar la preocupación de las mediciones, en primera instancia y en segunda instancia, para que los datos allí consignados sirvan de base para el control del equipo a lo largo del tiempo de uso. Hecho esto, no debiera olvidarse colocar alguna etiqueta con la indicación “CALIBRADO” y demás indicaciones al efecto. Claramente, siempre que se contrate un laboratorio acreditado para este tipo de tarea, habrá que cerciorarse que la incertidumbre de medición que pueda alcanzarse sea la adecuada para el uso previsto del instrumento a calibrar.

Por otra parte, el período que transcurrirá entre una calibración y la siguiente será determinado por el usuario del equipo – o de la balanza, en nuestro caso – en función de la frecuencia de uso, observación de su desempeño, trato brindado al equipo, etc. Suele ser habitual que las balanzas sean calibradas al menos una vez al año. Esto no es un criterio absoluto, ya que debe tenerse en cuenta que la calibración es necesaria cuando se quiere tener control sobre los errores e incertidumbres de la balanza para brindar evidencia objetiva de la conformidad del producto con los requisitos determinados.

5.3 Patrones

Las actividades que ha realizado el laboratorio de calibración acreditado, fueron realizadas con patrones reconocidos que aseguran la trazabilidad a patrones nacionales. Ahora bien, el laboratorio que encargó el trabajo, puede también contar con patrones para el control de sus equipos a los que llamamos “patrón de trabajo”. Según el Vocabulario Internacional de Metrología vigente, un Patrón de trabajo es un Patrón utilizado habitualmente para verificar instrumentos o sistemas de medida. Advierte además dos cosas: por un lado, que el mismo se calibrará con relación a un patrón de referencia y por otro, que a ese patrón de trabajo utilizado en la verificación se lo designará también como “patrón de verificación (comprobación)” o “patrón de control”. En suma, con posterioridad a la calibración, el control del equipo que realizará el laboratorio que realiza las mediciones, será fuertemente dependiente del patrón de trabajo que disponga.

5.4 Verificación

Pero ¿de qué tipo de control hablamos? En este caso particular, hablamos de la “verificación”. Sucede que muchas veces, la confusión por los términos proviene de las dificultades que emergen de las traducciones que se realizan. Si se lee atentamente la Norma ISO/IEC 17025 vigente, en inglés, se pueden ver dos situaciones interesantes:

a. Que la palabra “check” ha sido traducida como “***verificación***”.
b. Que “cross-check” es traducida en algunos casos como “***comprobaciones** cruzadas”*.

Todo ello sigue sumando más palabras a la confusión. Por el contrario, el espíritu del presente trabajo es el de ocuparnos más de los conceptos y no tanto en el término que lo describe o en este caso que lo traduce. Veamos qué dice el Vocabulario Internacional de Metrología (el VIM de aquí en adelante) vigente

> “2.44 **Verificación**: Aportación de evidencia objetiva de que un elemento satisface los requisitos especificados.”

Y agrega:

> *“EJEMPLO 1 La confirmación de que un material de referencia declarado homogéneo lo es para el valor y el procedimiento de medida correspondientes, para muestras de masa de valor hasta 10 mg.*
>
> *EJEMPLO 2 La confirmación de que se satisfacen las propiedades de funcionamiento declaradas o los requisitos legales de un sistema de medida.*
>
> *EJEMPLO 3 La confirmación de que puede alcanzarse una incertidumbre objetivo.*
>
> *NOTA 1: Cuando sea necesario es conveniente tener en cuenta la incertidumbre de medida.*
>
> *NOTA 2: El elemento puede ser por ejemplo, un proceso, un procedimiento de medida, un material, un compuesto o un sistema de medida.*
>
> *NOTA 3: Los requisitos especificados, pueden ser por ejemplo, las especificaciones del fabricante.*

Nota 4: En metrología legal, la verificación tal como la define el VIM L y en general en la evaluación de la conformidad, puede conllevar el examen, marcado o emisión de un "documento" de verificación de un sistema de medida.

Nota 5: No debe confundirse la ***verificación*** *con la* ***calibración****. No toda verificación es una validación. …"*

En suma, el VIM, en su versión vigente, no deja dudas en cuanto a que una calibración no es lo mismo que una verificación y que ambas se distinguen si se trata de actividades de metrología legal o de evaluación de la conformidad.

Entonces, qué se quiere decir con ese término. Lo que se busca es que se lo identifique con la idea de realizar actividades durante el período que transcurre entre dos **calibraciones**, lo que daremos en llamar "**verificación intermedia**". Algo que la Norma se ocupa de definir como sigue:

"5.6.3.3 ***Verificaciones intermedias****: Se deben llevar a cabo las verificaciones que sean necesarias para mantener la confianza en el estado de calibración de los patrones de referencia, primarios, de transferencia o de trabajo y de los materiales de referencia de acuerdo con procedimientos y un programa definidos"*

Sumando más confusión aún, puede apreciarse aquí que no hay consistencia terminológica ya que se habla de verificación intermedia en relación a los patrones y no a un equipo. No obstante, rescatamos la idea de que el laboratorio debe contar con un programa definido para llevar a cabo actividades de verificación **de la calibración** como así también los procedimientos apropiados al efecto. Para ello y volviendo al caso de nuestra balanza analítica, el laboratorio deberá contar con distintos patrones de trabajo (pesas) que representen cada uno de ellos, un solo valor o punto de control. Lo que se procura es determinar con ellos, si la balanza se encuentra dentro de los límites de error máximo permitido, a efectos de satisfacer los criterios de evaluación de la conformidad o verificación metrológica, además de otras características metrológicas como la estabilidad del instrumento, por ejemplo:

"4.26 Error máximo permitido (tolerado): valor extremo del error de medida, con respecto a un valor de referencia conocido, permitido por especificaciones o reglamentaciones, para una medición, instrumento o sistema de medida dado."

Y agrega:

Nota 1: En general, los términos "errores máximos permitidos" o "límites de error" se utilizan cuando existen valores extremos.

Nota 2: No es conveniente utilizar el término «tolerancia» par designar el error máximo permitido.

Lo habitual es que en una **verificación intermedia** se controlen puntos de medición de interés para el operador del laboratorio, que correspondan en el orden de las mediciones realizadas en los intervalos involucrados, en relación con ciertas especificaciones o requerimientos del cliente. Hecho esto, no debiera olvidarse colocar alguna etiqueta con la indicación "VERIFICADO" y demás indicaciones al efecto.

Así el objetivo de la **verificación intermedia** es conocer si el instrumento se encuentra dentro de los límites de error máximo permitido, obtenido durante la **calibración**.

5.5 Cronograma de Verificación

Toda estrategia diseñada para llevar a cabo verificaciones intermedias deberá tener en cuenta los costos involucrados, por lo que el responsable del laboratorio deberá asegurarse que el mínimo de valores obtenidos le otorgará verdadera confianza en los resultados. Sin duda, para ello, habrá considerado la última calibración del instrumento, el mensurando, factores de influencia, exactitud encontrada en valor de puntos de control, incertidumbre del patrón, etc.

Supongamos un laboratorio que utiliza la balanza para determinar el peso de las muestras, entre otras actividades de pretratamiento, antes de la medición. Conforme a lo expresado en 5. b. se ha establecido la rutina de realizar una calibración anual siempre que mantenga el flujo de trabajo de 100 determinaciones mensuales (dm). Ello en virtud de estudios estadísticos realizados por el laboratorio. Pero ¿cuál sería un cronograma apropiado de verificaciones a realizar durante el tiempo que transcurra entre ambas verificaciones? ¿Cuántas verificaciones intermedias debieran realizarse para que la actividad sea eficaz y eficiente? ¿Y que sucede si aumenta o disminuye la frecuencia de mediciones?

Aquí entonces entran a jugar no sólo la frecuencia, sino todas las consideraciones antedichas. De manera entonces que habrá que definir algún algoritmo que indique, en función de la actividad mensual del laboratorio, cada cuántos días sería apropiado realizar las verificaciones intermedias de la calibración. Si se mantiene la frecuencia normal (FN) antedicha de 100 dm (algo más de tres por día) cualquier balanza podría soportar dicho trabajo sin verificaciones intermedias durante tres meses. Siempre y cuando haya sido operada por personal competente y no hayan ocurrido incidentes o accidentes que le hayan producido un daño visible. Así, supongamos que el laboratorio definió 90 días como el período normal de control (PNC) del equipo para esa frecuencia. De esta manera:

FN establecida ⇒ una verificación intermedia cada 90 días

En el Anexo 1, se puede observa lo dicho en una gráfica. Ahora bien, ¿qué pasa si la cantidad de determinaciones mensuales (Cdm) se duplica, triplica o se reduce a la mitad? ¿se conservará el PNC?. Pues el sentido común dicta que habrá que variar el período de control (PC). En ese habrá que empezar por algo bien sencillo como lo que sigue e ir controlando si esa solución satisface al laboratorio:

$$\textbf{PC (días) = PNC x FN/Cdm}$$

Veamos esto con ejemplos:

Ejemplo 1: Suponemos que se ha reducido a la mitad la FN, esto es, ahora tenemos 50 dm. Resulta:

$$\textbf{PC (días) = 90 x 100/50 = 180 días}$$

Ejemplo 2: Suponemos ahora que se ha triplicado la FN, esto es, ahora tenemos 300 dm. Resulta:

$$\textbf{PC (días) = 90 x 100/300 = 30 días}$$

El registro de los datos obtenidos en las verificaciones intermedias resultará muy útil a la hora del análisis del desempeño del equipo. De allí que se sugiera el uso de cartas de control o programas apropiados como el de control estadístico de procesos (SPC por su sigla en inglés) o el Handbook of Statistical Methods.

Como resultado de todo lo anterior observamos que lo dicho hasta aquí contribuye a satisfacer los requisitos de la Norma en los puntos 5.5.2; 5.5.5. c); 5.5.10; 5.6.1; 5.6.3.3; y si se ha sido cuidadoso en la selección del Laboratorio que brinde el servicio de calibración, se habrá cumplido también con el punto C.5.6.2.1 de las Criterios del OAA.

Ahora bien, a pesar de este esfuerzo, no se habrá cumplido con el requisito 5.5.9 de la Norma, toda vez que luego de la calibración efectuada por el Laboratorio contratado para la realización de ese servicio, el equipo ha estado *fuera del control directo del laboratorio*. Por lo tanto, para cumplir con este punto de la Norma, el laboratorio deberá *asegurarse que se comprueban el funcionamiento y el estado de calibración del equipo y de que son satisfactorios, antes de que el equipo sea reintegrado al servicio.*

Ello implica realizar una recepción controlada del equipo que supone, en primera instancia, un análisis exhaustivo del Certificado de Calibración recibido (lo que requeriría un análisis exhaustivo que no se puede

hacer acá) y más tarde, una verificación de que cumple las especificaciones o requisitos definidos en los procedimientos técnicos en uso. Veamos lo que dice la Norma al respecto:
"4.6. Compras de servicios y suministros
4.6.1 El laboratorio debe tener una política y procedimientos para ***la compra de servicios*** *y suministros que utiliza …"*

Y agrega:

"4.6.2 El laboratorio debe asegurarse que los suministros comprados, que afectan la calidad de sus ensayos o de las calibraciones, ***no sean utilizados hasta que no hayan sido inspeccionados o verificados*** *de alguna forma como que cumplen las especificaciones normalizadas o los requisitos definidos en los métodos relativos a los ensayos o calibraciones concernientes."*

Es decir, será necesario documentar con todo detalle la verificación realizada, luego de la recepción del equipo que ha estado fuera del control del laboratorio, para demostrar que se ha dado cumplimiento a lo requerido en la Norma antes de ponerlo en operación para realizar la primera medición.

6. CONCLUSIONES

De todo lo anterior, se desprende que al hablar de calibración y verificación de equipos de laboratorio nos referimos a actividades diferentes. Asimismo, una verificación, no reemplaza a una calibración. Un laboratorio que no cuente con un cronograma de verificaciones intermedias apropiadas a la frecuencia de uso de sus equipos, no tendrá cómo demostrar la confianza en sus resultados. La calibración de la que hasta aquí se ha hablado, no deberá confundirse con los test operativos o de funcionamiento, en inglés autotest o self-calibration, que suelen llevar a cabo de manera automática en algunos equipos.

La puesta en práctica de las actividades aquí descriptas, permitirá al laboratorio, encontrar el justo equilibrio entre trabajo y costos para determinar su cronograma de verificación de equipos, evitar la aparición de resultados no conformes, aumentar el conocimiento del desempeño de sus equipos, demostrar a los clientes - con evidencias objetivas - la calidad de las mediciones y además cumplir con los puntos antes enumerados de la Norma y los Criterios.

Por otra parte, no debemos olvidar que detrás de cada ensayo, hay un ciudadano que está esperando que se le brinde un servicio de calidad. Todos consumimos alimentos, usamos equipos alimentados con energía eléctrica, nos hacemos análisis clínicos, usamos productos químicos, extintores presurizados, etc. Por lo tanto, todo laboratorio que preste atención a estos conceptos aumentará la confianza de la sociedad en sus resultados al tiempo que contribuirá a mejorar la calidad de vida de sus integrantes.

7. REFERENCIAS BIBLIOGRÁFICAS

Criterios Generales para la Evaluación y Acreditación de Laboratorios de Ensayo / Calibración (2009)

International Vocabulary of Metrology—Basic and General Concepts and Associated Terms (2008).

International Vocabulary of Terms in Legal Metrology (2000).

IRAM 301:2005 ISO/IEC 17025. (2005) Requisitos Generales para la Competencia de los Laboratorios de Ensayo y de Calibración.

ISO/IEC 17025. (2005) General Requirements for the Competence of Testing and Calibration Laboratories.

NIST/SEMATECH. e-Handbook of Statistical Methods.

SPC. Statistical Process Control. (2005).

Vocabulario Internacional de Metrología — Conceptos Fundamentales y Generales y Términos Asociados (2008).

Norm Survey in Argentina

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NORM SURVEY IN ARGENTINA

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Abstract- A survey programme was initiated several years ago with the aim to estimate NORM exposure incidence on workers in several different activities carried out through the country, in oil and gas industry, gold mining, spas and in caverns. This work presents the procedures, methods employed and the results found up to date from the survey including protection and remedial actions recommended when deemed necessary. Radium isotope concentrations measured in some samples were well above the exemption values established by the Standards. Elevated radon levels (above action level value established for workplaces) were detected in gas facilities, in the gold mine and cavern. The values detected were informed to the pertinent authorities as well as the facilities in order to take actions to reduce concentrations below the action level. In relation with the spas, almost all values of the geothermal waters analysed were below the corresponding guidance levels. 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. Some regulatory aspects for the management of NORM are suggested.

Keywords: NORM, non nuclear-industries.

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 (Hipkin et ál. 1991, Radiation Protection 107 1999, Reed et ál. 1991).

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

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^{210}Pb, which usually forms a thin cap in the internal surface of processing equipment (Radiation Protection 95 1999, Smith et ál. 1996). 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 the case of underground mines and caverns radon gas may concentrate up to high levels, particularly in the exploration stage. The exposure of workers by inhalation of radon gas may be significant (IAEA 2006).

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 (Radolic et ál. 2005). Then it may be possible to find out significant radon levels at places like geothermal spas. 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 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 related with gas and oil industry were characterized, one underground gold mine, one tourist cavern and 10 thermal spas.

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. Some regulatory aspects for the management of NORM are suggested.

2. INSTALLATION DESCRIPTIONS

2.1. Oil installations

2.1.1. Installation 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. Installations B, C and D

These installations 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.

Once the tubes arrive, they 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 installations

2.2.1. Installation 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 de-ethanizing tower retains ethane at the top, a de-propanizing tower retains propane at the top and 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. Installations F and G

These two installations 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

An underground mine for the extraction of gold and silver was surveyed. At the moment radon gas measurements were performed, the installation was in the exploitation step and evaluating the ventilation system efficiency in some galleries.

2.4. Tourist cavern

A particular karst tourist cavern of 1343 meters whose sedimentation environments were developed during the Jurassic and Cretaceous periods was surveyed. It has wide calcite deposits, consisting mainly of stalagmites and stalactites. It presents three levels of main corridors. The third corridor is constituted by alleys that descend 20 m below the first one.

2.5. Thermal spas recreation places and health resorts

A total of 10 thermal spas were evaluated. The first assessed spa was of volcanic origin of its geothermal water (Thermal Spa 1). It is being evaluated since 2005 to determine its evolution over time. Since 2007 a group of nine spas that uses water coming from aquifer systems (groundwater reservoirs) started to be monitored (Thermal Spa 2 to 10).

3. MEASUREMENTS

The relevant industry sectors were identified in each case. The first priority was to focus on the type of operations identified from current knowledge and experience as being the most likely to require attention (IAEA 2006). In situ measurements were performed and also sampling and analyses were conducted to determine activity concentrations and radon gas measurements.

In the case of oil and gas installations, to perform the identification of the industry sectors that could have been contaminated with NORM, 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 installations radon gas was also measured in the different separation streams.

In the case of the underground mine and cavern, radon gas measurements in air were carried out at different locations inside these installations, including all the galleries.

Radon detectors were placed at different points of each thermal spa and also equilibrium factors were measured. Water samples were taken for analysis of natural radionuclides to ARN laboratories.

3.1. In situ measurements

3.1.1. Oil and gas installations

Dose rate measurements were carried out in predetermined areas in agreement with the processes performed in each facility. The equipment used were Scintillation detector NaI(Tl) Identifinder and Geiger- Müller detector Automess.

First of all background measurements were performed in the surroundings of each installation. Then, in installations 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 installations A, E, F and G.

Installation	Background measurement (μSv/h)	Range of dose rate values (μSv/h)	Number of measurements in range
A	0.20 ± 0.02	Background level	9
		< 2	5
		2 - 10	9
		10 - 20	2
		> 20 (28.2 and 30)	2
E	0.10 ± 0.02	Background	7
		< 1	6
F	0.15 ± 0.04	Background	9
		< 2	11
		2 - 10	5
		> 10 [a]	5
G	0.12 ± 0.03	Background	19
		< 1	11
		1 - 3	16

[a] See table 2 for details

In Table 2 values above 10 μSv/h found in F installation are presented separately:

Table 2: Dose rates exceeding 10 μSv/h at F installation.

Sampling points	Dose rate in contact (μSv/h)	Dose rate at 1 meter (μSv/h)	Dose rate at 3 meters (μSv/h)
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 installations 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 installations B, C and D.

Installation	Background measurement (μSv/h)	Points above background	Dose rate values in contact (μSv/h)
B	0.09 ± 0.01	1	2.2
C	0.11 ± 0.01	0	-
D Store area	0.13 ± 0.01	1	2.8
D Washing area [a]		3	1-10
		1	10-20

[a] See table 4 for details

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

Table 4: Measurements in the washing area at installation D

Sampling points	Dose rate values in contact (μSv/h)	Dose rate values at 1 meter (μSv/h)	Dose rate values at 3 meters (μSv/h)
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	-
API vessel	0.100 – 0.130	-	-

3.1.2. Tourist cavern

12 different sampling points were measured along the cavern with an Automess 6150 detector. All the results were below 0.1 μSv/h.

3.1.3. Thermal spas

Dose rate measurements performed at Thermal Spa 1 are shown in Table 5. Background measurements were performed in the surroundings of the spa and were within natural radiation levels (0.1 – 0.2 μSv/h).

Table 5: Dose rate measurements at Thermal Spa 1.

Description	Dose rate (μSv/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
Sorroundings 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

3.2. ARN laboratory measurements

3.2.1. Oil and Gas installations

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

First, the samples were analysed by gamma spectrometry using Canberra GeHp detectors, with 30 % of efficiency. Then, Ra-226 analyses were performed by a radiochemical method, based on the co-precipitation of radium with $BaSO_4$ 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 6.

Table 6: Maximum and minimum radium isotopes and natural uranium concentration values in samples from installations A, B, C and D

Facility	Uranium		^{226}Ra		^{228}Ra	
	Minimum value	Maximum value	Minimum value (Bq/g)	Maximum value (Bq/g)	Minimum value (Bq/g)	Maximum value (Bq/g)
A	< 0.4 µg/g	1.9 ± 0.8 µg/g	< 0.1	1270 ± 130	115 ± 11	1670 ± 17
B	< 10.0 µg/L	33.0±9,8 µg/L	< 1.7 E-3	26.8 ± 2.7	< 1.1 E-3	9.6 ± 0.9
C	< 10.0 µg/L	1.5 ± 0.7 µg/g	< 1.4 E-3	0.07 ± 0.01	< 9.6 E-4	0.1 ± 0.01
D	< 0.4 µg/g	< 0.7µg/g	1.9E-3 ± 4E-4	18.7 ± 1.8	2.1E-3 ± 4E-4	65.4 ± 6.5

In E, F and G installations, radon gas measurements were performed by Lucas cell method (Lucas 1957). This method consists in collecting air samples in cells coated with SZn(Ag) and then the cells were measured in Ludlum 2200 alpha counters. The results are presented in Table 7.

Table 7: Radon gas concentrations in the different gas streams at installations E, F and G

Installation	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%)

3.2.2. Underground mine

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 (Canoba et ál. 1999). The results are presented in Table 8.

Table 8: Radon gas concentrations in air of the underground mine surveyed

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

*Uncertainty: 10% with K=2

3.2.3. Tourist cavern

In the case of the tourist cavern surveyed, radon gas in air was firstly measured by activated charcoal as screening and then by time-integrated detectors, CR-39 and Makrofol track detectors, (Urban 1986, Al-Najjar et ál. 1989, Dörschel et ál. 1994). Equilibrium factor (F) between radon and its daughters was also measured (Khan et ál. 1993, Canoba et ál. 2003). The results can be seen in Table 9.

Table 9: Radon gas concentrations in air of the tourist cavern surveyed

Sampling Points	Activated charcoal (Bq/m^3)	Track etched detectors* (Bq/m^3)
A	2831 ± 258	2297
B	963 ± 88	761
C	1222 ± 111	1494
D	1184 ± 108	903
E	---	1317
F	1062 ± 97	1168
G	1256 ± 115	1084
H	---	1126
I	427 ± 39	424
J	923 ± 84	1017
K	405 ± 37	792
L	409 ± 37	707
LL	250 ± 23	482
M	219 ± 20	435
N (Entry)	116 ± 11	321

The equilibrium factor measured was between 0.3 and 0.6 /*Uncertainty: 20% with K=2

3.2.4. Thermal spas

In order to characterize waters, two water 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. Radon in water was performed by liquid scintillation (ARN 2009). Uranium concentration was measured by fluorimetry or by KPA (Kinetic Phosphorescence Analysis) (ASTM 2002). Ra-226 and Pb-210 determinations were performed by radiochemical methods based on their precipitation as sulfates, and final measurement by liquid scintillation (Canoba et ál. 2005). Results are shown in Tables 10 and 11.

Table 10. [Rn-222] (Bq/m^3) in geothermal waters used for medical purposes at Thermal spas

Thermal spa	Description	Minimum value	Maximum value
1	Water at spring 1	4500 ± 1400	8700 ± 2000
	Water at spring 2	3300 ± 1100	15600 ± 2700
	Water at spring 3	< 98	5400 ± 1400
	North spring	5752 ± 1500	9000 ± 2000
	South spring	4200 ± 1200	11000 ± 2400
	Outdoor bath 1	5100 ± 1000	28900 ± 6100
	Outdoor bath 2	3600± 800	29800 ± 6300
	Ferrous water at spring	< 98	1900 ± 700
	Ferrous water in bath	-	2636 ± 571
	Sulphurous water at spring	862 ± 238	7400 ± 1800
	Sulphurous water outdoor bath	3327 ± 708	5500 ± 1400
	Sulphurous water indoor bath	2518 ± 574	9100 ± 1900
	Volcano water	< 98	10900 ± 2400
2	Water at spring	< 1000	-
3	Water at spring	< 1000	-
4	Water at spring	2168± 461	5230 ± 1094
5	Water at spring	2036 ± 435	2132 ± 461
6	Water at spring	< 1000	2563 ± 549
7	Water at spring	1835 ± 391	2625 ± 560
8	Water at spring	5249± 1094	5957 ± 1246
9	Water at spring	2273 ± 489	3388 ± 712

Table 11. [Uranium], [Ra-226] and [Pb-210] in geothermal waters

Uranium		Ra-226		Pb-210	
Samples below LD	Values above LD (μg/L)	Samples below LD	Values above LD (Bq/L)	Samples below LD	Values above LD (Bq/L)
22 (LD: 0.1 μg/L) n = 41	Minimum 0.14 ± 0.01 Maximum 28.4 ± 2.8	29 (LD: 0.01 Bq/L) n = 46	Minimum 0.02 ± 0.03 Maximum 1.13 ± 0.10	41 (LD: 0.06 Bq/L) n = 43	Minimum 0.07 ± 0.02 Maximum 0.17 ± 0.03

LD: detection limit

In order to measure radon levels in air, radon detectors were placed at different locations of each thermal spa. In the case of thermal spa 1, the spa is constructed in a fumarole area. The results regarding thermal spa 1 are shown in Tables 12.

Table 12. [Rn-222] in air at Thermal Spa 1

Sampling points	[Rn-222] (Bq/m^3) Activated charcoal	[Rn-222] (Bq/m^3) Time integrated detectors
Health office	140 ± 15	120 ± 25
Bath A	1543 ± 170	1100 ± 200
Office bath A	490 ± 50	373 ± 70
Corridor bath B	300 ± 30	68 ± 15
Bath C	205 ± 20	414 ± 80
Corridor bath C	305 ± 30	254 ± 50
Bath D	770 ± 80	*
Bath D, sulphurous water	861± 90	877 ± 160
Corridor bath D	177 ± 20	113 ± 20
Bath E	854 ± 90	1755 ± 340
Corridor bath F	600 ± 60	459 ± 90

*Detectors lost

In table 13 are shown radon levels in air at thermal spas 2 to 10.

Table 13: [Rn-222] in air (Bq/m^3) at thermal spas 2 to 10, with CR-39 detectors.

Thermal spa	Sampling location	[Rn-222] (Bq/m^3)*
2	Jacuzzi	199
	Emergency room	84
	Indoor swimming pool	87
3	Emergency room	35
	Indoor swimming pool	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

*Uncertainty: 20% with K=2

4. DISCUSSION

4.1. Oil and gas facilities

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 μSv/h, 15% were in the range 2–10 μSv/h and 9% were above 10 μSv/h.

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 14:

Table 14. Results of external exposure assessments in oil and gas facilities

Facility	Pieces above background	Maximum dose rate (μSv/h)	Occupational factor (hours/y)	Annual effective dose (mSv/y)
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]	320	0.26
		18.5[c]	25 (5 minutes per day - 300 days in a year) in contact	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 mSv/y
[b] Dose rate at 1 meter. /[c] Dose rate in contact.

In oil facilities, an annual effective dose of 0.6 mSv/y 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 mSv/y. It is suggested to optimize the doses received by workers in these areas 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 mSv/y. 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.

The incorporation of radiactive material is an exposure pathway that becomes important during washing and maintenance processes, in which workers may intake by inhalation particulate material. It was no 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 gas facilities that radon gas is concentrated in ethane and propane streams at very high levels. 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.

The analyses performed by fluorimetry in the samples analysed in the laboratory showed that uranium is not concentrated in sludges and scales. 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 the Standards (IAEA, 2004), namely 1 Bq/g for uranium and thorium series radionuclides, irrespective of the quantity of material and whether is in its natural state or has been subject to some form of processing.

4.2. 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 internationally agreed reference level value established for workplaces 1000 Bq/m^3, considered to be used globally in the interest of international harmonization (ICRP 2007, IAEA 1996). The radon concentration is very influenced by factors related to the entry of radon into the air (rock porosity) and removal of radon from the air (ventilation conditions). It was suggested to verify the ventilation system effectiveness.

4.3. Tourist cavern

From the radon gas measurements performed inside the cavern it can be seen that, in the sampling points named A, C, E, F, G, H and J, the values resulted above the corresponding reference level value established for workplaces (ICRP 2007, IAEA 1996, IAEA 2003). It was suggested to optimize the spent time of tourist guides.

4.4. Thermal spas

Similar radon gas results were obtained with both types of detectors in Spa 1. In order to assess the maximum dose that a worker may receive from radon gas inhalation, it was taken into account the highest value measured, 1755 Bq/m^3, at Thermal Spa 1. Although this value is above the corresponding reference level value, as the spent time of workers is well known (people work in this place only six months per year because the rest of the year it is covered with snow), the annual effective dose was calculated with a spent time of 1000 hours. The annual effective dose calculated was 6 mSv/y. The individual dose criteria established for deriving reference radon levels, both for members of the public and for workers is 10 mSv/y (ICRP 2007, ICRP 1993b). The maximum annual effective dose calculated resulted below this reference dose criterion.

In relation with water characterization, almost all values obtained for the geothermal waters analyzed were below the corresponding guidance levels recommended by WHO for drinking waters: 100.000 Bq/m^3 for Rn-222, 0.1 Bq/l for Pb-210 and 1 Bq/l for Ra-

226 (WHO 2011). Only two values were above the corresponding guidelines values: one value of Pb-210 of 0.17 Bq/l and one value of Ra-226 of 1.13 Bq/l. Although these waters are not used as drinking waters, the annual effective dose by ingestion was calculated for both radionuclides in a conservative way, considering an annual ingestion of 730 liters and each corresponding dosimetric factors (ICRP 1996). For Pb-210, the annual effective dose calculated was 0.08 mSv/a, and 0.23 mSv/a for Ra-226. Both values are well below the dose limit for members of the public, 1 mSv/a, (ICRP 2008). For natural uranium, in order to calculate the maximum result obtained (28.4 µg/L) in terms of activity concentration for U-238, the corresponding specific activity and abundance for U-238 in natural uranium were used (NPL 2008). The result was 0.34 Bq/L of U-238, well below the corresponding WHO guideline value (10 Bq/L).

5. CONCLUSIONS AND RECOMMENDATIONS

The dose rates measured at most oil facility locations were within background levels. Some points which resulted above background were from tubing with NORM and from washing and maintenance areas. The wastes are stored in each facility until removal by services companies. An annual effective dose conservatively estimated from the highest dose rate measured was 0.93 mSv/y. It is suggested that the doses received by workers in these areas be reduced by examining the possibilities for optimising the occupancy times.

In gas facilities some dose rates resulted well above background in the ethane and propane flows. The values measured were higher in the older facility owing to a greater surface accumulation of radionuclides in the piping. The annual effective dose calculated from the highest dose rate value measured was 1.6 mSv. Although the time spent by workers reported from the staff in the highest dose area is short, it was suggested that the presence of workers in these areas has to be justified and the situation properly informed to the personnel.

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.

For those pieces with dose rate measurements above background it would be important to define the suitable storage methods. As some pieces are sold as scrap 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 (API 1992, IAEA 2003).

Although the annual effective doses calculated (external exposure) are below dose limit established for workers (ICRP 2008), clear and open procedures for optimization of protection for the management of NORM are advisable.

In some scale and sludge samples analysed, radium isotopes were well above exemption values. In spite of this, the scenarios analyzed implied that the calculated doses were well below the dose limit established for workers. In this sense it is important to say that protective strategies would be implemented in relation with the characteristics of the exposure situations.

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 that locations of the mine and cavern surveyed where radon measurements resulted above the reference level established for workplaces, actions were recommended in order not only to reduce radon concentration values below the reference level but also to assure that protection has been optimized. In these environments that are conducive to the buildup of radon in air, particularly in underground sites, exposure of workers is the principal cause of concern. In the case of the mine, it was recommended to the facility to improve all the ventilation system. In the case of the cavern it has to be ensured that activity concentrations of ^{222}Rn in the workplace are below the reference level, and protection is optimized. Both exposure situations have to be kept under review.

The maximum annual effective dose calculated in the thermal spas from radon inhalation resulted below the reference dose criteria. In relation with spa water characterization, almost all values obtained for the geothermal waters analyzed were below the corresponding guidance levels recommended by WHO for drinking waters.

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

Finally, we believe that the management of NORM situations has to be subjected to a graded approach consistent with the optimization principle (ICRP 2007). Regulatory requirements cannot be based on a generic application of reference levels because of the large variety of processes, materials and activity concentrations, even within a same industry. A case by case analysis is recommended. The costs and benefits of introducing regulatory requirements also need to be considered and compared with other options that would achieve the same objective.

6. REFERENCES

Al-Najjar, S., Oliveira, A., Piesch, E., 1989. Radiation Protection Dosimetry, 27, p. 5.

API,1992. American Petroleum Institute, Bulletin on Management of Naturally Occurring Radioactive Materials in Oil & Gas Production, (Bul E2), API Official Publication, Dallas, USA.

ARN, 2009. Manual de Procedimientos de la Gerencia de Apoyo Científico Técnico, Determinación de trazas de uranio en agua con el equipo KPA-Acreditado por el OAA.

ASTM, 2002. Norma D-5174-02.

Canoba, A. C., López, F. O., Oliveira, A.A., 1999. Radon determination by activated charcoal adsorption and liquid scintillation measurement, Journal of Radioanalytical and Nuclear Chemistry, Vol. 240 Nº1, p.237-241.

Canoba, A.C., López, F.O., 2003. Passive method for the determination of the equilibrium factor between ^{222}Rn gas and its short period progeny, Journal of Radioanalytical and Nuclear Chemistry, Vol. 258 Nº2, p.269-274.

Canoba, A.C., Gnoni, G.A, 2005. Método simple para la determinación de Pb-210 y Ra-226 en aguas. Presented in the X Congreso de la Sociedad Española de Protección Radiológica, Huelva, España.

Dörschel, B., Piesch, E., 1994. Radiation Protection Dosimetry., 54, p. 41.

Hipkin, J., Payner, R.A., 1991. “Radiation Exposures to the Workforce from Naturally Occurring Radioactivity in Industrial Processes”, Radiation Protection Dosimetry, Vol. 36, N° 2/4, p.97-100.

IAEA, 1996. International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources. Safety Series 115. International Atomic Energy Agency, Vienna, Austria.

IAEA, 2003. Radiation Protection against Radon in Workplaces other than Mines, Safety Report Series N° 33, International Atomic Energy Agency (IAEA), Vienna, Austria.

IAEA, 2003. Safety Reports Series N° 34 Radiation Protection and Management of Radioactive Waste in the Oil and Gas Industry, Vienna, Austria.

IAEA, 2004. Aplication of the Concepts of Exclusion, Exemption and Clearence. IAEA Safety Standars Series, N° RS-G-1.7, Vienna, Austria.

IAEA, 2006. Safety Reports Series N° 49. Assesing the Need for Radiation Protection Measures in Work Involving Minerals and Raw Materials, Vienna, Austria.

ICRP, 1993b. Protection against radon-222 at home and at work. ICRP Publication 65. Ann. ICRP 23 (2).

ICRP, 1996. Age-dependent doses to the members of the public from intake of radionuclides. Part 5. Compilation of ingestion and inhalation coefficients. ICRP Publication 72. Annals of the ICRP, 26(1).

ICRP, 2007. Scope of radiological protection control measures. ICRP Publication 104. Annals of the ICRP 37 (5).

ICRP, 2007. The 2007 recommendations of the International Commission on Radiological Protection. ICRP Publication 103. Annals of the ICRP, 37 (2–4).

Khan A., Qureshi I., Tufail, M., 1993. Passive Dosimetry of Radon and its Daughters Using Solid State Nuclear Track Detectors (SSNTDs), Radiation Protection Dosimetry, Vol. 46 Nº3, p. 149-170.

Lucas, H. L., 1957. Rev. Sci. Inst., 28, 680.

NPL, 2008. National Physical Laboratory, NPL REPORT IR 6, Recommended Nuclear Decay Data, January 2008.

Radiation Protection 95, 1999. Reference levels for wokplaces processing materials with enhanced levels of naturally occurring radionuclides, Office for Official Publications of the European Communities, Luxembourg, European Commission.

Radiation Protection 107, 1999. Establishment of reference levels for regulatory control of workplaces where materials are processed which contain enhanced levels of naturally-occurring radionuclides, Office for Official Publications of the European Communities, Luxembourg European Commission.

Radolic, V., Vukovic, B., Smith G., et ál., 2005. Journal of Environmental Radioactivity 83, 191-198.

Reed, G., Holland, B., Mc Arthur, A., 1991. “Evaluating the Real Risks of Radioactive Scale in Oil and Gas Production”, First International Conference on Health Safety and Environment, El Hague, The Netherlands, 10-14 Nov. 1991, SPE 23383**,** p. 549-558.

Smith, K.P., Williams, G.P., Tebes. C.L., 1996. “Radiological dose assessment related to management of naturally occurring radioactive materials generated by the petroleum industry”, Environmental Assessment Division, Argonne National laboratory, Illinois, USA.

Urban, M., 1986. Nuclear Tracks, 12, p. 685.

WHO, 2011. Guidelines for Drinking-water Quality, fourth edition, Geneva, World Health Organization.

Jornada de actualización en dosimetría interna para el ciclo de combustible

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Argentina

INTRODUCCIÓN

El cálculo de la dosis efectiva, en los casos de exposición interna de trabajadores, requiere de la evaluación de la actividad incorporada a partir del análisis de datos de medición utilizando modelos biocinéticos y dosimétricos. En el caso de la exposición a uranio existe una dificultad adicional en el análisis de los datos de monitoreo debido a la excreción de uranio natural proveniente de la dieta.

El objetivo de este trabajo es presentar los contenidos y el balance de estas Jornadas realizadas en el marco de las funciones de control y asesoramiento por parte de la Autoridad Regulatoria Nuclear a responsables de áreas controladas.

Dada la importancia de un abordaje específico en estos temas, se convocó a todos los involucrados en el control de la exposición interna de los trabajadores del ciclo de combustible a participar para discutir aspectos de interés sobre los planes de monitoreo y las herramientas de cálculo disponibles.

Esta actividad conjunta entre los Departamentos Control de Instalaciones del Ciclo de Combustible y Evaluaciones Dosimétricas y Radiobiológicas se desarrolló entre el 23 y el 24 de noviembre de 2010 y se realizó tomando como base el curso de Métodos Avanzados de Dosimetría Interna organizado en agosto de 2009 y enfocando las particularidades de la dosimetría interna para uranio.

AREAS TEMÁTICAS

Los temas abordados se centralizaron en la normativa vigente, en los criterios para el monitoreo de los trabajadores y en las herramientas para el cálculo de dosis específica para el ciclo de combustible, tomando en cuenta los diferentes compuestos de uranio y enriquecimientos. Las presentaciones en estas Jornadas, abordaron temas seleccionados para contribuir a la actualización que se detallan a continuación junto con los especialistas a cargo:

- "Ciclo del combustible nuclear", Hugo Plaza
- "Criterios para el monitoreo de la exposición interna"[1], Ana Rojo
- "Metodología para la evaluación de datos de medición: estimación de incorporación y dosis efectiva comprometida.", Inés Gómez Parada
- "Aplicación práctica de la publicación NRPB W22: Industrial Uranium Compounds: Exposure Limits, Assessment of Intake and Toxicity after Inhalation" [3], Nancy Puerta
- "Evaluación de un caso de inhalación de uranio natural"[2], Sebastián Gossio
- "Toxicidad química renal" [3], Ana Rojo
- "Normativa vigente" [4,5,6], Analía Saavedra

DESARROLLO DEL TEMARIO

El cronograma de actividades fue desarrollado con la participación en carácter de expositores, de personal de los Departamentos Control de Instalaciones del Ciclo de Combustible (Hugo Plaza y Analía Saavedra) y Evaluaciones Dosimétricas y Radiobiológicas (Ana Rojo, Inés Gómez Parada, Sebastián Gossio y Nancy Puerta) de la Autoridad Regulatoria Nuclear de Argentina,

Al inicio de las actividades se dio la oportunidad a los representantes de cada instalación para que presentaran los programas de monitoreo para el control de la incorporación de uranio de sus respectivas áreas y esto fue el marco adecuado para repasar la normativa vigente, las recomendaciones recientes de la ICRP[2], los nuevos estándares de la ISO[7,8] y publicaciones específicas de referencia de la HPA[3]. Cada responsable realizó una presentación sobre la "Descripción de aspectos relativos al plan de monitoreo de la exposición interna" en la que se detallaron los siguientes puntos:

- Compuestos de uranio involucrados: Tipo y enriquecimiento.
- Breve descripción de los tipos de tareas con potencial exposición interna: vía seca / húmeda. Cantidad de material manipulado por proceso y anualmente.
- Sistemas de protección operativos para evitar la incorporación: ventilación, uso de barbijos, etc.
- Planes de monitoreo vigentes para rutina y accidentes: personales y de área. Tipos de muestras y frecuencias de muestreo.
- Laboratorio responsable de las mediciones: técnicas y límites de detección
- Número de trabajadores involucrados.
- Registros disponibles: tipo de datos, evaluaciones realizadas.

Cabe señalar que en estas Jornadas se presentó una aplicación práctica de la publicación NRPB W22 en forma de planilla Excel, desarrollada por personal del Departamento de Evaluaciones Dosimétricas y Radiobiológicas que fue parte del material que se distribuyó a los participantes. Esta hoja de cálculo permite evaluar de forma rápida, datos de medición (directas o indirectas) provenientes del monitoreo individual para obtener la dosis efectiva comprometida, E(50), resultante en un trabajador.

La planilla permite seleccionar entre tres tipos de compuestos: U_3O_8, $U0_2$ y $U0_3$, considerando los parámetros de absorción propuestos por el NRPB-W 22. Esta planilla en primera instancia solo tiene como tipo de composición Uranio natural.

El usuario debe cargar los datos de medición de orina en µg/l o la carga en pulmón expresada en Bq, y el software calcula la E(50).

Además permite calcular la carga retenida en riñón indicando si hay riesgo toxicológico (se considera que existe riesgo toxicológico si la retención en riñón es mayor a 3 µg/g de riñón).

Se debe aclarar que los valores obtenidos mediante esta planilla, deberán ser analizados por un especialista en Dosimétrica Interna para su utilización en evaluaciones para registro de dosis.

La figura 1 muestra una pantalla de la aplicación mostrando a modo de ejemplo un caso de inhalación de uranio natural.

PARTICIPANTES

La convocatoria fue dirigida a personal de instalaciones del ciclo de combustible con conocimientos en dosimetría interna y preferentemente que hayan completado el curso de Métodos Avanzados dictado por ARN en 2009. Se contó con la presencia de personal de DIOXITEK, CONUAR y CNEA (CAE, CAC, CAB y el laboratorio de dosimetría personal -DPA-). La distribución de participantes se presenta en la siguiente tabla:

Nombre de la INSTALACIÓN	Número de PARTICIPANTES
CNEA USCAB	1
CNEA ECRI	2
CNEA LUE -LTA	4
CNEA Unidad de Seguridad	2
CNEA PFPU	1
CNEA - DPA	2
ARN- GACT	1
ARN- GSRFyS	1
DIOXITEK	1
CONUAR	3
TOTAL	18

CONCLUSIONES

El resultado de esta reunión con todos los involucrados en el control de la exposición interna de los trabajadores del ciclo de combustible fue muy satisfactorio, para reguladores y regulados. Se tuvo la posibilidad de analizar en detalle los programas de monitoreo de cada instalación para el control de la incorporación de uranio y esto permitió identificar y discutir coincidencias y discrepancias de interés apuntando a la mejora de la protección radiológica. Así mismo, permitió actualizar herramientas y unificar los criterios para la implementación de planes de monitoreo en concordancia con las últimas recomendaciones internacionales.

Se acordó dar continuidad en el futuro a estas jornadas para crear un espacio de discusión y de actualización de aspectos relativos a las mediciones y evaluaciones de los datos provenientes de los planes de monitoreo de las distintas instalaciones del Ciclo de Combustible, generando así un compromiso mutuo en un proceso de mejora continua de la protección radiológica ocupacional.

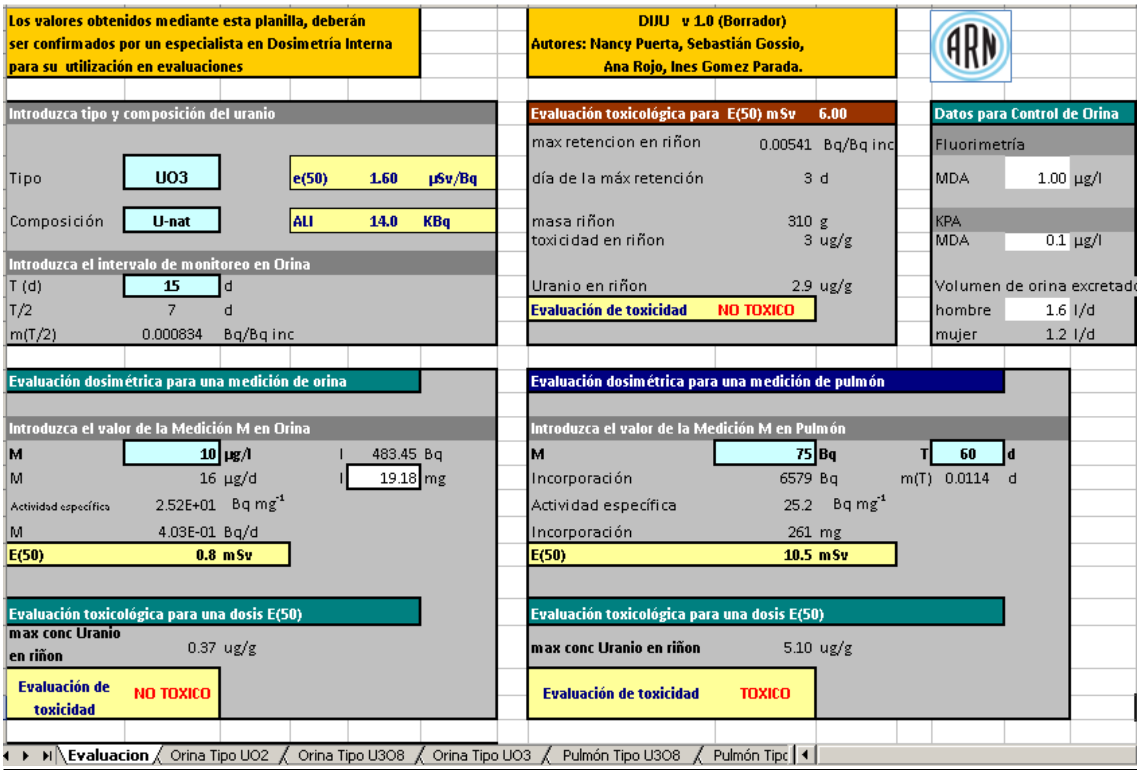
REFERENCIAS

1. ARN PI-4/07.

   Programas para el monitoreo de la exposición interna: criterios para su diseño. Rojo A; Gómez Parada, I.

2. Guías IDEAS. 2006 FZKA 7243.
3. NRPB W22: Industrial Uranium Compounds: Exposure Limits, Assessment of Intake and Toxicity alter Inhalation.
4. Norma básica de seguridad radiológica. AR 10.1.1 Revisión 3, año 2001.
5. Exposición Ocupacional en instalaciones radiactivas clase I. AR 6.1.1 Rev. 1 año 2002.
6. Limitación de Efluentes radiactivos en instalaciones radiactivas clase I. AR 6.1.2 Rev. 2 año 2002.
7. ISO 20553. 2006. Radiation protection — Monitoring of workers occupationally exposed to a risk of internal contamination with radioactive material.
8. ISO 27048. Radiation protection dose assessment for internal exposure.

Figura 1. Aplicación práctica de la publicación NRPB W22 en forma de hoja de cálculo Excel, desarrollada por personal del Departamento de Evaluaciones Dosimétricas y Radiobiológicas.



WORKSHOP ON INTERNAL DOSIMETRY IN THE NUCLEAR FUEL CYCLE

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Dose assessment in case of internal exposure involves the estimation of committed effective dose based on the interpretation of bioassay measurement, and the assumptions of hypotheses on the characteristics of the radioactive material and the time pattern and the pathway of intake. In the case of workers exposed in nuclear fuel facilities, the normal uranium excretion from the diet is an additional difficulty in the process of assessing internal exposure.

The aim of this paper is to present the main topics discussion and the conclusions of the workshop, held in the frame of the missions of the Autoridad Regulatoria Nuclear.

All the personnel involved in the control of internal exposure in nuclear fuel cycle was invited to participate in the workshop to discuss about individual monitoring criteria and the available tools for assessing committed effective dose in the workers of their facilities.

The lectures were presented jointly by the Nuclear Fuel Cycle Facilities Control and the Dosimetric and Radiobiological Assessment departments. It was hold at the Ezeiza Atomic Center from 23th to 24th November 2010 based on the Advanced Course on Internal Dosimetry organized on 2009 and focusing specific uranium compound internal dosimetry.

A representative of each facility was invited to present the monitoring program implemented for controlling the internal exposure. It was an opportunity to discuss criteria and to share experiences on this field in the frame of the ICRP, HPA and ISO publications. The different monitoring program criteria could be analyzed and so contributing to the improvement of radiological protection.

Finally, it was agreed to hold periodical meetings to assure the update on uranium measurement techniques and the handling of monitoring data for committed effective dose assessment.

Twenty Years of Regional Safeguards: the ABACC System and the Synergy with the National Nuclear Material Control Systems

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Vicens, H.E.; Maceiras, E. and Terigi, G.E.

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Twenty Years of Regional Safeguards: the ABACC System and the Synergy with the National Nuclear Material Control Systems

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Abstract
As result of the nuclear integration between Brazil and Argentina, in July 1991 the Agreement for Peaceful Uses of the Nuclear Energy (Bilateral Agreement) was signed and the Brazilian Argentine Agency for Accountancy and Control of Nuclear Material (ABACC) was created [1]. The main role assigned to ABACC was the implementation and administration of the regional control system and the coordination with the International Atomic Energy Agency (IAEA) in order to apply safeguards to all nuclear material in all nuclear activities of Argentina and Brazil. In December 1991 the IAEA, ABACC, Argentina and Brazil signed the Quadripartite Agreement (INFCIRC/435) [2]. The agreement establishes obligations similar to those established by model INFCIRC/153 comprehensive agreements. The Bilateral Agreement establishes that the Parties should make available financial and technical capabilities to support ABACC activities. In order to accomplish this challenge, the National Systems had to improve their structure and capabilities. Through the close interaction with the IAEA and ABACC, the national systems have been enriched by adopting new methodologies, implementing innovative safeguards approaches and providing specialized training to the regional inspectors. All of this also resulted in relevant technical improvements to the regional system as a whole. The approach of both neighborhoods controlling each other increased the confidence between the partners and permitted a better knowledge of their potentialities. The recognized performance of the regional system in the implementation of innovative, efficient and credible safeguards measures increased the confidence of the international community on the implementation of nuclear safeguards in Argentina and Brazil. In this paper, after twenty years of the creation of the ABACC System, the view of the Brazilian and Argentine National Authorities is presented.

Introduction

Research initiatives in the nuclear area developed by Brazil and Argentina, especially in the 80's and with involvement of military institutions, have aroused international suspicion that the countries were seeking to develop nuclear weapons. Realizing this negative atmosphere, both countries agreed to take actions in an attempt to demonstrate to the international community that the use of nuclear energy was intended only for peaceful applications. In 1980 the countries signed the Cooperation Agreement for Development and Application of the Peaceful Uses of Nuclear Energy [3]. In the same decade, several joint declarations on its nuclear policies were announced by the countries. In 1988, Brazil decided to include the same principle in its constitution and to attribute to the Congress the responsibility for approving any nuclear activity to be carried out in the Brazilian territory. Later, the countries decided to strengthen the approximation in the nuclear area through the creation of a Permanent Committee on Nuclear Policy, which adopted the scientific-technical cooperation as its main pillar and was aimed primarily to demonstrate transparency in their nuclear programs. However, the most significant initiative in this direction occurred in 1991 when the two countries signed and ratified the Agreement for Exclusively Peaceful Use of Nuclear Energy (Bilateral Agreement), which established a Common System for Accounting and Control of Nuclear Materials (SCCC), created ABACC and implemented a system of mutual inspections between them. The IAEA safeguards in the two countries had been applied since the 60's through specific agreements, but the establishment of the SCCC and the creation of ABACC conducted to the signature of a new agreement between Argentina, Brazil, ABACC and the IAEA in 1994 - the so-called "Quadripatite Agreement". Soon after, in May 1994, the Treaty on the Prohibition of Nuclear Weapons in Latin America (Tratelolco Treath) [4] entered into force for Brazil, and in September 1998 the country adhered to the Treaty on Non-proliferation of Nuclear Weapons (NPT) [5]. Argentina ratified its adherence to the dealings of Tlatelolco in January 1994 and to the NPT in February 1995.

The SCCC and the structure of ABACC

The SCCC, established by the Bilateral Agreement, has the goal to verify that all nuclear materials used in all nuclear activities in both countries are not diverted to nuclear weapons or other nuclear explosive devices. ABACC is an international organization that aims to administrate the implementation of the comprehensive safeguards system (SCCC) established by the Bilateral Agreement. ABACC has two bodies: the Commission and the Secretariat. The Commission of ABACC is comprised of four members, two from Argentina and two from Brazil, all appointed by the respective governments. The Commission approves the regulations for operation of ABACC, supervises the work of the Secretariat, approves its technical officers and informs of any abnormalities to the corresponding Party, which will be required to take all the necessary actions to remedy the situation. The Secretariat of ABACC is formed by a Secretary and a Deputy Secretary, with nationalities alternating each year, a group of eight Technical Officers (four from each country), administrative staff and approximately 90 inspectors nominated by the countries and approved by the Commission. The inspectors are usually specialists who perform activities in the nuclear area in their respective countries as members of the National Authorities or as employees of other official organizations of the nuclear area. Therefore, they are inspectors who work for ABACC in a temporary basis, upon request of the Secretariat. Inspections carried out in Brazil are necessarily performed by inspectors from Argentina and vice versa. It should be noted that the Technical Officers of the Secretariat, who are permanent staff of ABACC, can also perform inspections. This scheme assigns a unique feature to the SCCC in terms of implementation of nuclear safeguards in the world.

The Quadripartite Agreement

As stated earlier, both Argentina and Brazil were already signatories of safeguards agreements with the IAEA when the SCCC was established and ABACC created. Immediately after the signing of the Bilateral Agreement, in the same month of December 1991, Argentina, Brazil, ABACC and the IAEA signed the Quadripartite Agreement for the joint application of safeguards by the IAEA and ABACC in both countries. The Agreement considers that each agency shall reach its own conclusion in terms of accomplishment of the Agreement by the Parties, while avoiding unnecessary duplication of efforts. The Agreement entered into force in March 1994 and represented a milestone in the international regime of nuclear non-proliferation by presenting a model of bilateral application of comprehensive safeguards. The Quadripartite Agreement was prepared with the same principle that the NPT in regards to application of comprehensive IAEA safeguards, although the countries had not yet signed such a

Treaty. The obligations of the countries under the Quadripartite Agreement are well known, as they are similar to those valid for signatories of model INFCIRC/153 agreements with the IAEA.

The National Nuclear Material Control Systems in Argentina and Brazil
In Brazil, the government's efforts for establishing an independent system of control of nuclear materials became more evident in the early 80s. At that time, the Brazilian government was implementing a standalone program to develop nuclear technology in order to acquire appropriate capacity to use nuclear energy according to the needs of the country. The Nuclear Energy Commission of Brazil (CNEN) was the Brazilian government organization responsible for performing such a control, a situation that persists today. To meet its commitments at the national level, CNEN established an appropriate structure, including technical and laboratorial support, capable of performing a domestic control totally independently of the international control applied by the IAEA so far. CNEN is empowered to issue regulatory documents that establish requirements for nuclear material control by operators of nuclear installations, as well as to issue authorizations for the use and transfer of such material.

From 1956 to 1994, the nuclear regulatory function in Argentina was carried out by the “Atomic Energy Commission”. In 1994, the Argentine government decided to split into three different organizations the management of the generation and the use of nuclear energy from the regulation of such activities. Since then, the regulatory activities have been carried out by the Nuclear Regulatory Authority (ARN).

In this framework, the Nuclear Regulatory Authority (ARN) has, among its functions, to ensure that the nuclear material and activities are not deviated to any unauthorized purpose and that they are performed in compliance with all the international undertakings and commitments assumed by Argentina.

Since 1994, the ARN has developed a complete set of measures and procedures, called National Standard, that are applicable to all nuclear materials, equipment, installations and materials of nuclear interest. These National Standards empowered ARN with all the attributes of an independent State System of Accounting and Control (SSAC) that regulates all the nuclear activities within the State at national level.

The infrastructure for nuclear material control existing in Argentina and Brazil at the time of the creation of ABACC was fundamental for its establishment in the first phase, especially with regard to human capital, as officials from the governments of Argentina and Brazil who were permanent staff of the National Authorities were nominated by their governments to form part of the Secretariat of ABACC. A similar situation occurred with respect to the structure for training and some equipment used in inspections, which were lent to ABACC until it could acquire their own equipment.

With the signing of the Quadripartite Agreement and sometime after the start of the joint inspections regime of the IAEA and ABACC in both countries, the role of National Authorities gained even more importance, especially because, according to the Bilateral Agreement, the National Authorities should provide the necessary support to ABACC so that it could fulfill its mission successfully.

Impact of the Regional Safeguards System on the Activities of the National Authorities

The unique characteristics associated with the SCCC in terms of application of safeguards caused significant structural and operational impact on the activities performed by the National Authorities of Argentina and Brazil. The main points that have suffered this influence are presented below:

- Temporary Inspectors: This is perhaps one of the main particularities the SCCC. ABACC inspectors are permanent staff of the National Authorities or other official organizations of the nuclear area. Some of them routinely act as operators of nuclear facilities (power and research reactor, fuel fabrication and conversion plants, laboratories, etc.) and, therefore, undergo routine inspections of both the IAEA and ABACC. This variety of tasks offers inspectors the opportunity to better understand both sides of the system, to be constantly informed of advances in both areas and, especially, understand the importance of implementing adequate control of nuclear materials and apply these concepts more efficiently in their routine work. The application of domestic safeguards tends to be improved when there are operators who are also temporary inspectors of ABACC. In the case of the staff of the National Authorities, experience as an inspector of the regional system contributes significantly to a better implementation of its routine work due to the more diverse capacitation in-deep knowledge about different processes, technologies and types of nuclear facilities that, sometimes, do not exist in their own country. To ABACC, there is a distinct gain to have inspectors who understand well the various types of facilities (operators) and have the experience to perform daily tasks associated with control and accountancy of nuclear material (national inspectors).

- Communications, Information Exchange and Coordination of Activities: According to the Quadripartite Agreement, ABACC concentrates almost all of the information related to international safeguards applied in Argentina and Brazil. This makes ABACC a critical point in the flow of information. However, while the IAEA deals with information related to dozens of countries, ABACC deals only with two. Typically, this allows greater agility in handling and analyzing the information not only for ABACC, but also for the States. The discussion forums foreseen in the SCCC (bilateral and trilateral meetings) allow better flexibility in negotiations until a certain level. In addition, the logistics related to meetings between Argentina, Brazil and ABACC within the SCCC is facilitated by the proximity between the neighbors and the offices of ABACC. The work of ABACC in planning and coordination of the activities to be performed in the two countries improves the process of analysis, approval and implementation of such activities.

- Infrastructure: The responsibility assigned by the Bilateral Agreement to National Systems of Argentina and Brazil in providing the necessary support to ABACC to perform its mission effectively, meant that countries had to constitute adequate material and human resources so that they could meet this dual function of conducting the domestic control and meeting possible demands from ABACC. As a result, both National Authorities maintain a wide range of equipment, materials, computer programs and other tools used for physical verification and accountability of nuclear materials. The infrastructure for laboratory analysis in the area of safeguards in both countries has also suffered a strong influence, since ABACC uses it for obtaining results of analysis of samples collected by its inspectors.

- Training: This point of influence is strongly related to the topic “temporary inspectors”. Inspectors of the National Authorities who are also inspectors for ABACC receive additional periodic training in various types of activities they need to perform as inspectors of the regional system. The workouts are specific, often provided by high-level experts, and cover a wide variety of procedures, equipment and techniques. The knowledge obtained during these trainings usually has direct impact on their daily activities as staff of the National Authorities or operators. Moreover, several of the training equipment of ABACC is available to inspectors for training in non-routine basis, in case this need is appropriately identified.

Current and Future Challenges

The growing up of the nuclear activities in Argentina and Brazil has imposed to ABACC and the respective National Authorities additional challenges in regards to limited financial and human resources to accomplish the same objectives. The organizations have to seek for more efficient methods and approaches in order to save resources. The know-how in nuclear safeguards is usually concentrated and takes 5 to 10 years for being transferred to new generations. The National Authorities of Argentina and Brazil have to dedicate special attention to this problem and consider that ABACC could play an important role is this challenge.

Conclusions

After twenty years of the SCCC and ABACC, Argentina and Brazil have experienced significant changes in the way external safeguards is applied for both countries. The existence of the regional system brought into the game the atmosphere of mutual confidence that nuclear energy is used only for peaceful purposes. The close interaction with ABACC allows the States to deal with safeguards related topics in a routine basis, optimizing the implementation process. Improvements in the capacity to apply domestic safeguards were also noticed by the respective National Authorities, since the Bilateral agreement requires that the States have the adequate capacity to support ABACC whenever necessary. Argentina and Brazil believe that safeguards can be much more efficient if the capabilities of the state systems are taken fully into consideration and a well-established regional system is recognized as an indicator of fully accomplishment of international safeguards commitments by the Parties.

References

[1] Agreement Between The Republic of Argentina and the Federative Republic of Brazil for the Exclusively Peaceful Use of Nuclear Energy, INFCIRC/395, International Atomic Energy Agency, November 1991.

[2] Agreement Between the Republic of Argentina and the Federative Republic of Brazil, the Brazilian-Argentine Agency for Accounting and Control of Nuclear Materials and the International Atomic Energy Agency for the Application of Safeguards, INFCIRC/435, International Atomic Energy Agency, March 1994.

[3] Cooperation Agreement for Development and Application of the Peaceful Uses of Nuclear Energy, Official Publication of the Brazilian Government , n^{o} 215, November 1983.

[4] Treaty for the Prohibition of Nuclear Weapons in Latin America and the Caribbean (Tlatelolco Treath), February 1967.

[5] Treaty on the Non-proliferation of Nuclear Weapons, INFCIRC/140, International Atomic Energy Agency, April 1970.

Determinación de uranio más plutonio por espectrometría alfa en varias matrices

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DETERMINACIÓN DE URANIO MÁS PLUTONIO POR ESPECTROMETRÍA ALFA EN VARIAS MATRICES

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RESUMEN

En general, la determinación de emisores alfa por espectrometría se lleva a cabo realizando una purificación previa de cada uno de los elementos que se desea cuantificar. En este trabajo se describe una metodología para la determinación de isótopos de uranio y plutonio en forma conjunta, con el propósito de mejorar los tiempos de procesamiento analítico y de medición. El método consiste en la purificación de uranio y plutonio, y la posterior electrodeposición para la medición por espectrometría alfa. La técnica se basa en el uso de TBP (tributilfosfato) como extractante, y reactivos de fácil obtención. Es aplicable a diversas matrices, entre ellas aguas, filtros y suelos. En las condiciones descriptas, se aplica a alícuotas pequeñas de aproximadamente 0,5 g de sólidos. La técnica produce electrodepósitos de alta calidad.

1. INTRODUCCIÓN

Para la determinación de emisores alfa por espectrometría, el criterio generalizado es realizar una purificación previa de cada uno de los elementos a cuantificar. Esto se debe a que los espectros presentan picos de baja resolución, por lo que, para evitar la superposición de picos y mejorar el análisis de las áreas, es conveniente tener un bajo número de picos en el espectro.

Para la medición de U y Pu en forma conjunta por espectrometría alfa es necesario lograr varias condiciones.

En primer lugar es indispensable obtener un electrodepósito de muy buena calidad que permita generar espectros con picos muy bien resueltos, teniendo en cuenta que hay picos cercanos en el espectro que podrían solaparse. Para ello es necesaria una purificación eficiente de estos elementos respecto de la matriz inactiva.

En segundo lugar, es necesario separar a aquellos emisores alfa que por tener energías de emisión cercanas a los isótopos del U o del Pu puedan comportarse como interferentes en el análisis espectral. Los emisores alfa mas importantes que interfieren en el análisis del U o Pu son: ^{241}Am, ^{210}Po, ^{230}Th, ^{228}Th y ^{237}Np.

El ^{241}Am debe ser separado del U y el Pu, ya que su presencia en el espectro interfiere con el ^{238}Pu, y la presencia de ^{243}Am (trazador del ^{241}Am) interfiere con el ^{232}U (trazador del U) y el ^{239}Pu.

En el caso del uranio, es condición indispensable que el ^{210}Po no este presente, dado que este se solapa completamente con el trazador de ^{232}U, es decir que la metodología debe asegurar la separación del ^{210}Po.

El torio es también una interferencia importante, tanto en lo que se refiere al solapamiento del ^{230}Th y el ^{229}Th (trazador del Th) con el ^{234}U, como en el solapamiento del ^{228}Th con el ^{238}Pu y de la energía 5340,4 keV con el ^{232}U. Como el ^{228}Th es la hija del ^{232}U y además crece relativamente rápido (T1/2 ^{228}Th: 698,6 d), es de esperar que siempre se lo encuentre presente en las soluciones de ^{232}U, en mayor o menor medida. Además, la energía de 5448,8 keV del ^{224}Ra, que crece a expensas del ^{228}Th (T1/2 ^{224}Ra: 3,627 d) interfiere con el ^{238}Pu. Por esto, la técnica debería asegurar la eliminación del Th.

El ^{237}Np es también una interferencia importante ya que, para la técnica propuesta tiene un comportamiento químico parecido al del U y Pu, e interfiere en el espectro puesto que tiene una energía similar a la del ^{234}U y cercana a la del ^{242}Pu.

Los isótopos ^{239}Pu y ^{240}Pu presentan energías muy similares, de tal forma que su cuantificación en forma individual no es posible en las condiciones habituales de trabajo, por lo que se informan como ^{239}Pu + ^{240}Pu.

La tabla 1 muestra las dos principales energías y las intensidades de los emisores alfa más comunes, donde se mencionan aquellos que constituyen interferencia para la medición de U + Pu.

Isótopo [1,2]		Energía (keV)	Intensidad (%)	Energía (keV)	Intensidad (%)
Th-232		4011,2	78,9	3948,5	21,0
U-238		4198	77,54	4151	22,33
Th-230 [3]	Interferencia	4687,0	76,3	4620,5	23,4
Th-229 [3]	Interferencia	4845,3	56,20	4901,0	10,20
U-234		4774,6	71,37	4722,4	28,42
Np-237	Interferencia	4788,0	47,64	4771,4	13,0
Pu-242		4902,3	76,53	4858,2	23,44
Pu-239		5156,59	70,79	5143,82	17,14
Pu-240		5168,13	72,74	5123,6	27,16
Am-243	Interferencia	5275,3	86,74	5233,3	11,46
Po-210	Interferencia	5304,33	99,999		
U-232		5320,24	69,1	5263,48	30,6
Th-228	Interferencia	5423,28	73,2	5340,38	26,2
Am-241	Interferencia	5485,56	84,45	5442,86	13,23
Pu-238		5499,03	71,04	5456,3	28,85
Ra-224	Interferencia	5685,48	94,73	5448,8	5,25
Cm-243		5786,4	73,4	5742,5	11,3
Cm-244		5804,77	76,7	5762,65	23,3

Tabla 1. Se muestran las dos principales energías e intensidades de los emisores alfa más comunes y su capacidad de comportarse como interferencias en un espectro de uranio más plutonio.

La técnica se basa en el uso de un extractante neutro, el TBP (tributilfosfato), de reconocida eficacia en la extracción de muchos cationes en diferentes medios, y reactivos de fácil obtención. La extracción de U y Pu se lleva a cabo en medio HCl 6 M, luego se realizan sucesivos lavados con HNO_3 6 M y finalmente se reextraen los isótopos de U y Pu neutralizando primero con amoniaco y luego reextrayendo con $(NH_4)_2CO_3$ 10% más ácido ascórbico para reducir el Pu a valencia III y facilitar su pasaje a la fase acuosa.

La técnica es rápida, pudiéndose procesar varias muestras en pocas horas de trabajo, dejando la medición en curso durante la noche. Por otro lado, es económica ya que utiliza reactivos comunes en pequeñas cantidades. Además, produce electrodepósitos de alta calidad.

Es aplicable a diversas matrices tales como aguas, filtros, suelos y sólidos diversos, que en general producen problemas a los analistas por su alto contenido de elementos activos e inactivos (entre ellos el Fe, Th, Po, Pa, etc.) y no requiere la implementación de una coprecipitación previa, que es una práctica usual para separar parte de la matriz. En las condiciones descriptas, se aplica a alícuotas pequeñas, estimativamente de 0,5 g de residuo. Su rango de aplicabilidad puede ampliarse si se usan ampollas de decantación, que permiten trabajar con volúmenes mayores que los viales de centelleo.

Como referencia, la tabla 2 presenta los coeficientes de distribución para varios emisores alfa entre HCl 6 M y TBP 100%, y entre HNO_3 6 M y TBP 100%.

Elemento	Kd (HCl 6 M)	Kd (HNO_3 6 M)
U (VI)	**80** ([4]-p142), **95** ([7]-p953, [6])	**300** ([7]-p953),**400** [6]
Pu (IV)	**40** [6]	**200** [6]
Th (IV)	**0,02** ([4]-p142, [5]-p16, [7]-p953, [6])	**90** ([5]-p17, [7]-p953), **300** [6]
Po		**0,03** ([7]-p953)
Am (III)		**0,2** ([4]-p138)
Pu (III)	**0,05**([7]-p953)*	
Cm (III)	**0,05**([7]-p953)*	**0,25** ([4]-p138)

* HCl 9M

Tabla 2. Valores de coeficiente de distribución (Kd) aproximados entre TBP 100% y HNO_3 6 M - HCl 6 M.

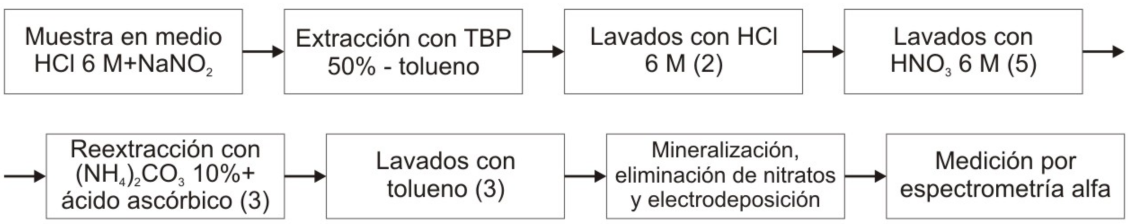
Además, la técnica debe asegurar la eliminación del hierro y demás elementos de la matriz inactiva. Como referencia, la tabla 3 presenta los coeficientes de distribución para varios cationes, componentes comunes de la matriz suelo o similares, entre HCl 6 M y TBP 100%, y entre NO_3H 6 M y TBP 100%.

Elemento	Kd (HCl 6 M)	Kd (HNO_3 6 M)
Fe (III)	**400** ([4]-143) **8000** ([[7]-955)	**0,01**([7]-953)
Zn (II)	**15**([4]-143), **10**([7]-955)	
Cu (II)	**0,3** ([4]-143), **1** ([7]-955)	**2** ([7]-953)
Co	**0,3** ([4]-143), **1** ([7]-955)	**0,008**([7]-953)
Mn (II)	**0,004** ([7]-955)	**0,003**([7]-953)
Pa	**180** ([7]-955)	**80**([7]-953)
Al	**0,005**([7]-955)	**0,003**([7]-953)

Tabla 3. Valores de coeficiente de distribución (Kd) aproximados entre TBP 100% y HNO_3 – HCl para algunas interferencias.

2. DIAGRAMA DE FLUJO

El diagrama de flujo de la técnica es el siguiente:



3. MATERIALES Y EQUIPOS

- TBP (tributilfosfato) 50% en tolueno.
- HCl 6 M p.a.
- $NaNO_2$ p.a.
- HNO_3 6 M p.a.
- $(NH_4)_2CO_3$ p.a.10% en agua.
- Ácido ascórbico p.a.
- H_2SO_4 concentrado p.a.

- Solución de electropulido (apéndice 1).
- Fuente de electrodeposición de corriente constante de 1 A.
- Centrífuga de mesa.
- Espectrómetro alfa con detectores de ion implantado.

4. DISOLUCIÓN DE SUELOS

En una bomba de teflón de 30 ml se pesan hasta 0,5 g de suelo o residuo, y se agregan los trazadores de ^{232}U y ^{242}Pu o ^{236}Pu. La disolución comprende 2 etapas, en la primera se agregan 9 ml de agua regia y se coloca en estufa con ventilación a 150 ^{o}C durante una hora, el líquido se pasa a un vial de centelleo, se centrifuga, se vuelca el sobrenadante en un vaso de teflón y se pone a evaporar; la parte no disuelta se pasa con 4 ml de HNO_3, 2 ml de H_2O_2 y 4 ml de HF a la misma bomba, la que se coloca en estufa otra hora más. El líquido se agrega al vaso de teflón y se continúa la evaporación hasta sequedad. Luego, se trata varias veces con porciones de HCl concentrado, llevando a sequedad cada vez.

5. PROCEDIMIENTO ANALÍTICO

- Se redisuelve el residuo con 5-10 ml de HCl 6 M trasvasando a un vial de centelleo de vidrio de 20 ml con varias porciones. Se agrega unos 100 mg de $NaNO_2$, se agita y se deja reposar 5 minutos. Si aparece precipitado, se centrifuga, lavando un par de veces con HCl 6 M. El $NaNO_2$ convierte al Pu presente en Pu IV. Si no se requiere la determinación de Pu, no es necesario el agregado de $NaNO_2$.
- Se agregan 7 ml de TBP 50% en tolueno y se agita por 2 minutos Se centrifuga unos segundos y se separa la fase orgánica (FO) con una pipeta Pasteur plástica transfiriendo a otro vial.
- Se repite el punto anterior.
- Se desecha la fase acuosa (FA) o se reserva eventualmente en un vial de centelleo para analizar Th, Am y Cm.
- Al vial conteniendo la FO se agregan 5 ml de HCl 6 M, se agita 1 minuto, se centrifuga unos segundos y se separa la FA, que se descarta o se adiciona a la anterior. La FA de HCl 6 M contiene Am, Th, trazas de Pu, y parte de la matriz. El Fe es transportado por la FO.
- Se repite el punto anterior. Con 2 lavados se elimina casi totalmente el Th.
- Al vial conteniendo la FO se agregan 5 ml de HNO_3 6 M, se agita 1 minuto, se centrifuga unos segundos y se separa la FA, la que se descarta.
- Se repite el punto anterior 4 veces más. Este lavado permite eliminar el Fe y además el Po. Los últimos 2 lavados deben ser con HNO_3 de buena calidad, es decir con bajo contenido de Fe.
- Al vial conteniendo la FO se agregan 1 ml de agua y se neutraliza con amoniaco y azul de timol como indicador (alrededor de 60 gotas). Luego se agregan 2 ml de $(NH_4)_2CO_3$ 10%, una punta de espátula de ácido ascórbico, se agita 1 minuto, se centrifuga unos segundos y se separa la FA, transfiriéndola a un vial. El ácido ascórbico convierte al Pu presente en Pu III. Si no se requiere la determinación de Pu, no es necesario el agregado de ácido ascórbico.
- Luego se agregan 5 ml de $(NH_4)_2CO_3$ 10%, una punta de espátula de ácido ascórbico, se agita 1 minuto, se centrifuga unos segundos y se separa la FA, transfiriéndola al vial.
- Se repite el punto anterior.
- Se centrifuga unos segundos el vial con los líquidos de reextración y se reparan los restos de TBP.
- Se agregan 3 ml de tolueno, se agita 30 segundos, se centrifuga unos segundos y se separa la FO.

- Se repite el punto anterior 2 veces más.
- Se trasvasa a un vaso de precipitados de 100-150 ml, se evapora suavemente hasta casi sequedad. Se mineraliza con HNO_3/H_2O_2. Se coloca el vaso a aproximadamente 300 ^oC hasta la eliminación de las sales amónicas.
- Se trata con HCl, luego clorhidrato de hidrazina y luego mayor temperatura para lograr la eliminación del cloruro polonioso. Se repite el procedimiento llevando luego a mufla a 350 ^oC, 30 minutos.
- Se repite el punto anterior 1-2 veces más.
- Enfriar, agregar cuidadosamente 0,5 ml de H_2SO_4, volver a la plancha y agregar H_2O_2 para eliminar los nitratos.
- Agregar 3 ml de agua destilada, ajustar el pH a 2 con azul de timol y electrodepositar (ver apéndice 1).

6. DISCUSIÓN

Los nitratos provenientes de la neutralización del TBP cargado con HNO_3, son eliminados con H_2O_2, luego del agregado de H_2SO_4. Este es un paso importante para la obtención de un depósito óptimo, con buena resolución y alto rendimiento.

El pH de la electrodeposición es importante ya que una acidez mayor redisuelve el depósito provocando un rendimiento menor.

La presencia de ^{210}Po en el depósito es una seria interferencia pues al solaparse completamente con el pico de ^{232}U, hace muy difícil su detección y puede dar origen a una subestimación del resultado. En una prueba, la recuperación de Po, para 5 lavados de HNO_3 6 M resultó ser de 1,4%. En la técnica presentada, se incluye un paso para la eliminación de Po que consiste en su reducción a valencia II y su conversión a $PoCl_2$, que es el compuesto mas volátil de Po (sublima a 190 ^oC [9]), el que luego se elimina llevándolo a 350 ^oC.

La extracción del U en las condiciones mencionadas es prácticamente total, no registrándose presencia del mismo en las aguas madres, en varias pruebas realizadas.

Los dos lavados con HCl 6 M producen una pérdida de U de aproximadamente 4,2%, mientras que los lavados con HNO_3 8 M mostraron pérdidas similares a las anteriores (para entre 3 y 5 lavados).

La reextracción con agua y agua levemente carbonatada produjo pobres resultados entre 4 y 5% de recuperación. Este valor subió hasta casi 60% cuando se neutralizó el TBP y se reextrajo con agua. Cuando se neutralizó el TBP, se reextrajo con agua y luego con $(NH_4)_2CO_3$ 10% se obtuvieron valores de recuperación de alrededor de 78%.

En parte, el Np es también extraído en estas condiciones y se lo encuentra en el electrodepósito por lo que interfiere con el uranio. En el caso en que hubiera presencia de Np se puede reducir la muestra con ácido ascórbico, y de este modo extraer solo el U. Luego llevar a sequedad las aguas madres, mineralizar, retomar y extraer el Pu y el Np de la misma forma que se lleva a cabo la extracción de U y Pu.

En el caso de ser necesario determinar Th, una posibilidad de trabajo, dentro de este contexto, seria tomar la fracción de HCl 6 M, llevarla a sequedad, cambiar el medio por HNO_3 6 M y extraer nuevamente con TBP.

7. RESULTADOS

El siguiente es un espectro típico de uranio y plutonio obtenido con esta técnica:

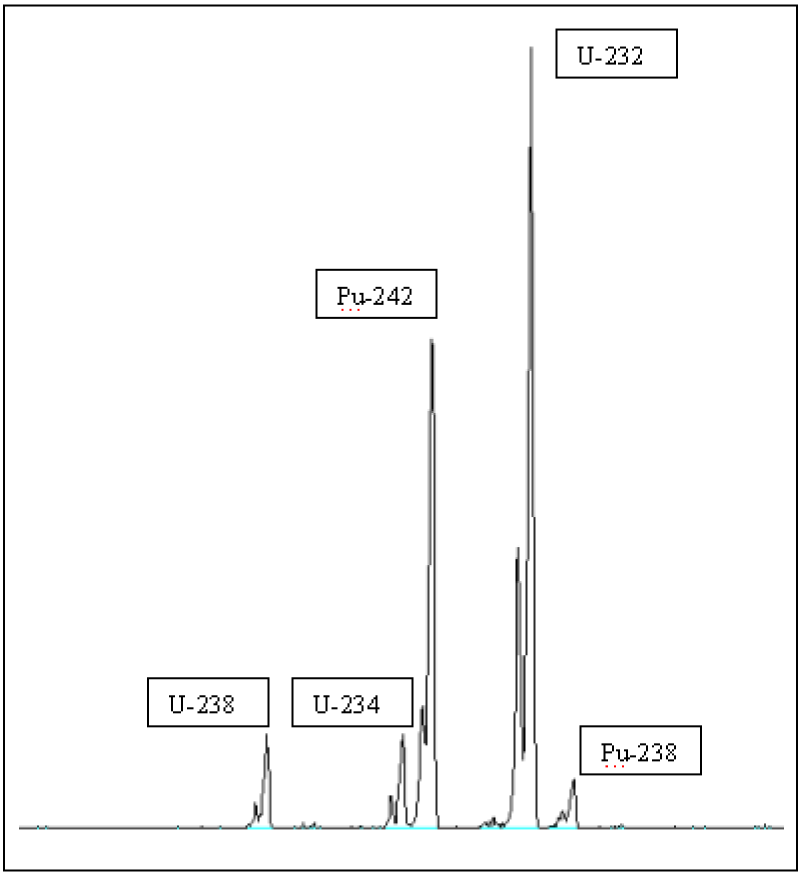


Figura 1. Espectro C042102A correspondiente a una de las determinaciones realizadas para la intercomparación CSN-2010 y obtenido con detectores PIPs.

En el espectro puede observarse claramente la separación entre los picos, especialmente entre ^{242}Pu y ^{234}U, cuyas energías son muy cercanas. Además, puede observarse una muy escasa prolongación de la cola característica de los picos alfa, relacionada con la degradación de la energía, así como también una marcada separación entre energías de un mismo radioisótopo.

La recuperación es alta, generalmente entre el 60% y el 90%.

8. CONCLUSIONES

La técnica descripta permite obtener resultados de isótopos de U e isótopos de Pu en una sola medición. Es aplicable a diversas matrices, tales como aguas, filtros y suelos. Es económica ya que utiliza reactivos comunes en pequeñas cantidades. Además, produce electrodepósitos de alta calidad, imprescindibles para un correcto análisis espectral. Por último, es una técnica rápida, que permite procesar varias muestras en pocas horas de trabajo.

9. REFERENCIAS

[1] Monographie BIPM-5, Table of Radionuclides, Vol 2, A=151-252, 2004.

[2] Monographie BIPM-5, Table of Radionuclides, Vol 4, A=133-252, 2008.

[3] http://laraweb.free.fr/ (acceso 01/03/11).

[4] NAS-NS 3050, The radiochemistry of uranium, 1972.

[5] NAS-NS 3004, The radiochemistry of Thorium, 1982.

[6] Journal of the Less-Common Metals, 122, 343 – 345, 1986.

[7] Ion Exchange and solvent extraction of metal complexes, Marcus and Kertes, 1969.

[8] Anal. Chem., 44, 2, 280, 1972.

[9] Handbook of Chemistry and Physics, CRC, 55th Edition, 1974-1975.

APÉNDICE 1
Electropulido y electrodeposición.

El método de electrodeposición utilizado fue el desarrollado por Talvitie en 1972 [8] y se lleva a cabo en viales de centelleo de plástico (Figura 2), sobre discos de acero inoxidable 316L ó 308 de 2 cm de diámetro y 1,6-1,7 cm de diámetro de depósito, con electropulido previo. El proceso de electropulido se realiza antes de electrodepositar y consiste en electrolizar 2-3 ml de una solución 50% de H_3PO_4, 50% de H_2SO_4, con corriente de 0,5 A invertida, en la misma celda de electrodeposición, durante 10 minutos Luego, el ánodo de Pt se lava hirviéndolo en una solución diluida de HCl a ebullición, y la celda se desarma y se lava con agua y agua destilada.

La electrodeposición se lleva a cabo en medio sulfato de amonio aproximadamente 1 M, a pH 2-2,5 y un volumen de aproximadamente 10 ml, con ánodo de Pt en espiral plano de 1 cm aprox. de diámetro, con 2-3 mm de distancia entre electrodos y 1 A de corriente continua constante, durante 2 hs. Se agrega 2-3 ml de NH_3 1:1 y se corta la corriente luego de unos segundos.

El esquema del sistema es:

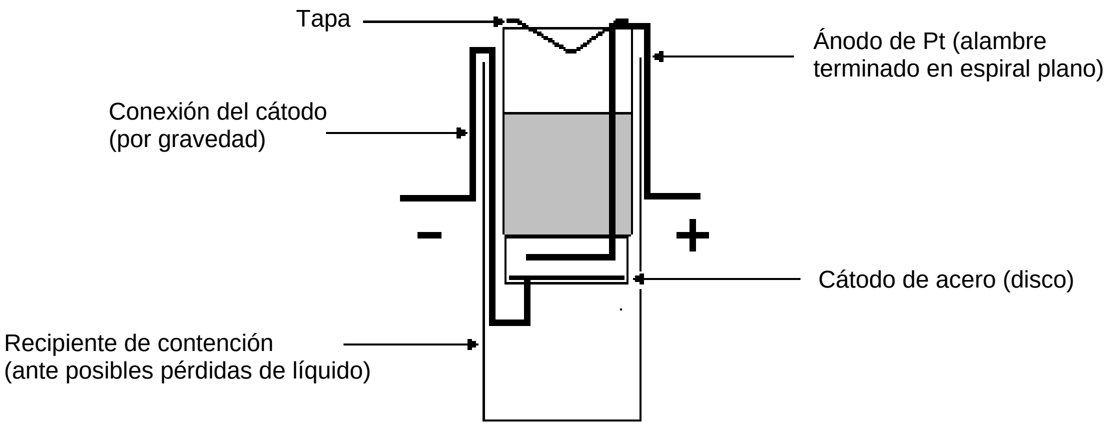


Figura 2. Esquema del sistema de electrodeposición.

PARTE II

Resúmenes de publicaciones en revistas

EXPERIMENTAL AND THEORETICAL COMPTON PROFILES OF Be, C AND Al*

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Argentina

Abstract

The results of Compton profile measurements, Fermi momentum determinations, and theoretical values obtained from a linear combination of Slater-type orbital (STO) for core electrons in beryllium; carbon and aluminium are presented. In addition, a ThomasFermi model is used to estimate the contribution of valence electrons to the Compton profile. Measurements were performed using monoenergetic photons of 59.54 keV provided by a low-intensity Am-241 γ-ray source. Scattered photons were detected at 90° from the beam direction using a p-type coaxial high-purity germanium detector (HPGe). The experimental results are in good agreement with theoretical calculations.

* Publicado en: Physica B: Condensed Matter. Volume 406, Issue 3, 1 February 2011, Pages 354-357.

BIOLOGICAL DOSIMETRY INTERCOMPARISON EXERCISE: AN EVALUATION OF TRIAGE AND ROUTINE MODE RESULTS BY ROBUST METHODS*

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[10] Instituto de Investigaciones Biológicas Clemente Estable, Uruguay
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Abstract

Well-defined protocols and quality management standards are indispensable for biological dosimetry laboratories. Participation in periodic proficiency testing by interlaboratory comparisons is also required. This harmonization is essential if a cooperative network is used to respond to a mass casualty event. Here we present an international intercomparison based on dicentric chromosome analysis for dose assessment performed in the framework of the IAEA Regional Latin American RLA/9/054 Project. The exercise involved 14 laboratories, 8 from Latin America and 6 from Europe. The performance of each laboratory and the reproducibility of the exercise were evaluated using robust methods described in ISO standards. The study was based on the analysis of slides from samples irradiated with 0.75 (DI) and 2.5 Gy (DII). Laboratories were required to score the frequency of dicentrics and convert them to estimated doses, using their own dose-effect curves, after the analysis of 50 or 100 cells (triage mode) and after conventional scoring of 500 cells or 100 dicentrics. In the conventional scoring, at both doses, all reported frequencies were considered as satisfactory, and two reported doses were considered as questionable. The analysis of the data dispersion among the dicentric frequencies and among doses indicated a better reproducibility for estimated doses (15.6% for DI and 8.8% for DII) than for frequencies (24.4% for DI and 11.4% for DII), expressed by the coefficient of variation. In the two triage modes, although robust analysis classified some reported frequencies or doses as unsatisfactory or questionable, all estimated doses were in agreement with the accepted error of ±0.5 Gy. However, at the DI dose and for 50 scored cells, 5 out of the 14 reported confidence intervals that included zero dose and could be interpreted as false negatives. This improved with 100 cells, where only one confidence interval included zero dose. At the DII dose, all estimations fell within ±0.5 Gy of the reference dose interval. The results obtained in this triage exercise indicated that it is better to report doses than frequencies. Overall, in both triage and conventional scoring modes, the laboratory performances were satisfactory for mutual cooperation purposes. These data reinforce the view that collaborative networking in the case of a mass casualty event can be successful.

* Publicado en: Radiation Research, Volume 175, Issue 5, May 2011, Pages 638-649.

COMPUTATION OF UNSTABLE FLOWS USING SYSTEM CODES*

Ferreri, J.C.

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Argentina

Abstract

The prediction of unstable flows by means of so-called systems-codes is discussed in this paper. In the nuclear field, verification and validation (V&V) of codes applicable to predict the behavior of complex nuclear installations is a matter of huge collective effort presently. Multi-Physics-Multiple-Scales models, coupled with the needed wall laws is just one example if length scales go into wall layers or minute geometrical details. The validity of physical models as applicable to situations of interest in developing nuclear technology systems (e.g. natural circulation systems, super-critical conditions flows, very high temperature gas-cooled reactors) is now under consideration in the technical literature. This even applies to very simple problems, like cavity flows or one-dimensional (1D) systems. Some basic problems are presented for V&V that must correctly be solved by any systems code. Based on this, conclusions are drawn in relation to some present systems-codes, regarding the sensitivity to code options influencing its numerical approach and physical closure correlations.

* Publicado en: Journal of Physics: Conference Series, Volume 296, Issue 1, 2011.

ENGINEERING JUDGMENT AND NATURAL CIRCULATION CALCULATIONS*

Ferreri, J.C.

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Abstract

The analysis performed to establish the validity of computer code results in the particular field of natural circulation flow stability calculations is presented in the light of usual engineering practice. The effects of discretization and closure correlations are discussed and some hints to avoid undesired mistakes in the evaluations performed are given. Additionally, the results are presented for an experiment relevant to the way in which a (small) number of skilled, nuclear safety analysts and researchers react when facing the solution of a natural circulation problem. These results may be also framed in the concept of Engineering Judgment and are potentially useful for Knowledge Management activities.

* Publicado en: Science and Technology of Nuclear Installations, Volume 2011, 2011.

NUCLEAR ACTIVITIES IN ARGENTINA, 2010*

Ferreri, J.C.[1,2]; Clausse, A.[2,3,4]; Ordóñez, J.P.[5] and Mazzantini, O.A.[6]

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[5] INVAP S.E.
[6] Nucleoeléctrica de Argentina (NA-SA)

Argentina

Abstract

An introduction is presented in which the editor discusses various reports published within the issue on topics including radiation safety, CANDU nuclear power plant, and nuclear reactor designs.

* Publicado en: Nuclear Activities in Argentina, 2010 Science and Technology of Nuclear Installations Volume 2011, 201

THE ARGENTINE APPROACH TO RADIATION SAFETY: ITS ETHICAL BASIS*

González, A.J.

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Argentina

Abstract

The ethical bases of Argentina's radiation safety approach are reviewed. The applied principles are those recommended and established internationally, namely: the principle of justification of decisions that alters the radiation exposure situation; the principle of optimization of protection and safety; the principle of individual protection for restricting possible inequitable outcomes of optimized safety; and the implicit principle of intergenerational prudence for protection future generations and the habitat. The principles are compared vis - vis the prevalent ethical doctrines: justification vis - vis teleology; optimization vis - vis utilitarianism; individual protection vis - vis deontology; and, intergenerational prudence vis - vis aretaicism (or virtuosity). The application of the principles and their ethics in Argentina is analysed. These principles are applied to ALL exposure to radiation harm; namely, to exposures to actual doses and to exposures to actual risk and potential doses, including those related to the safety of nuclear installations, and they are harmonized and applied in conjunction. It is concluded that building a bridge among all available ethical doctrines and applying it to radiation safety against actual doses and actual risk and potential doses is at the roots of the successful nuclear regulatory experience in Argentina.

* Publicado en: Science and Technology of Nuclear Installations, Volume 2011, 2011.

EPISTEMOLOGY ON THE ATTRIBUTION OF RADIATION RISKS AND EFFECTS TO LOW RADIATION DOSE EXPOSURE SITUATIONS*

González, A.J.
Nuclear Regulatory Authority

Abstract

The epistemology on the attribution of radiation risks and effects relates to the theories of knowledge applied for assigning harm to radiation exposure, especially with regard to methods, validity and scope. The attribution of radiation risk is associated with the concept of probability, including the plausibility of whether there is risk in prospective exposure situations. The attribution of radiation effects is based on the concept of provability, which involves demonstrability and counterfactuality and the attestability that actual effects have actually being incurred in retrospective exposure situations. Under present knowledge, radiation risks are attributable to low-dose radiation exposure situations, however small the doses may be. Thus, ascribing nominal radiation risks to prospective exposure situations for radiation protection purposes is required for reasons of duty, responsibility, prudence and precaution. However, the prospective attribution of radiation risk does not imply that actual effects can be automatically attributed to low-dose exposure situations.

* Publicado en: International Journal of Low Radiation, Vol. 8, No. 3, 2011.

METHOD FOR THE CALCULATION OF DPA IN THE REACTOR PRESSURE VESSEL OF ATUCHA II*

Mascitti, J.A. and Madariaga, M.

Nuclear Regulatory Authority

Abstract

One of the limiting factors of the life of a nuclear power plant (NPP) is the state of the reactor pressure vessel (RPV). Embrittlement is the most important effect affecting RPV aging. The irradiation with neutrons, especially fast neutrons, is the primary cause of this embrittlement. NPP safe operation requires to ensure RPV integrity over its lifetime, threatened by the neutron radiation-induced embrittlement. In this paper, we identify the areas where the RPV neutron radiation is maximum and perform calculations of the displacement-per-atom (DPA) rate in those areas using the MCNP5 code. It was determined that the maximum DPA rate in the RPV wall with fresh fuel element (FE) is 3.76(3) × $10^{-12}s^{-1}$, it takes place in front of FEs BA42 and BL43, and it is symmetrical about the central channel, LG04, and LH03.

* Publicado en: Science and Technology of Nuclear Installations, Volume 2011, 2011.

LA AUTORIDAD REGULATORIA NUCLEAR Y SU COMPROMISO CON LA ACREDITACIÓN*

Remedi, J.O

Autoridad Regulatoria Nuclear
Argentina

Resumen

El artículo informa acerca de los orígenes de la Autoridad Regulatoria Nuclear (ARN), como entidad autárquica y sus actividades vinculadas al compromise de proteger a las personas, el ambiente y las futuras generaciones de los efectos nocivos de las radiaciones ionizantes (RI) y de asegurar la naturaleza exclusivamente pacífica de los usos de la energía nuclear, basándose en la mejora continua de la eficacia de su Sistema de Gestión de la Calidad. La ARN se encuentra fuertemente identificada con la Calidad. En particular, mantiene su compromiso con la Acreditación habiendo alcanzado esa distinción para los laboratorios de ensayo y calibración que se detallan. Asimismo, todas sus actividades se encuentran cubiertas por un Sistema de Gestión de Calidad, aplicando el modelo de gestión de la serie de normas IRAM-ISO 9001:2008.

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RADIOLUMINESCENCE OF RARE-EARTH DOPED POTASSIUM YTTRIUM FLUORIDES CRYSTALS*

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Abstract

The radioluminescence (RL) properties of K_2YF_5 crystals doped with Ce^3+, Tb^3+ and Dy^3+ under ionizing irradiation excitation have been studied for the first time. The main objective of this work has been to assess the feasibility of using these crystals as detectors for fiberoptic radioluminescent dosimetry. In particular, it has been found that the RL intensity from both K_2YF_5:Tb (10%) and K_2TbF_5 is comparable to that from a commercial Al_2O_3:C crystal. Longer wavelength emission from these fluorides makes simple optical filtering technique possible to use in order to avoid the stem effect. Afterglow decay times for these fluorides have been found to be similar to that for Al_2O_3:C and, in particular, K_2TbF_5 does not show longer afterglow decay time compared to Al_2O_3:C

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THERMO- AND RADIOLUMINESCENCE OF UNDOPED AND DY-DOPED STRONTIUM BORATES PREPARED BY SOL-GEL METHOD*

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Abstract

Borates have shown to be efficient materials to be employed for ionizing radiation detection given their tissue-equivalent characteristics. Usually, the efficiency of these compounds is largely dependent on the preparation methods and conditions. In this work nominally pure and Dy-doped strontium borates have been prepared by sol-gel method. The thermoluminescence and radioluminescence of the resulting borates have been investigated and correlated to their structural characteristics.

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AN EFFICIENT ALGORITHM FOR COMPUTERIZED DECONVOLUTION OF THERMOLUMINESCENT GLOW CURVES*

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Abstract

The models employed so far for deconvolving thermoluminescent glow curves are either derived by neglecting the interaction among traps and resorting to the quasi-equilibrium approximation, or are simply phenomenological. Several published articles have shown that the approximations are difficult to justify. Further it has never been shown that they are rigorously applicable to any known system. As to the phenomenological model it is no physically meaningful. An algorithm is reported which allows analyses of glow curves without the aforementioned approximations.

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REFERENCE METHODOLOGIES FOR RADIOACTIVE CONTROLLED DISCHARGES AN ACTIVITY WITHIN THE IAEA'S PROGRAM ENVIRONMENTAL MODELLING FOR RADIATION SAFETY II (EMRAS II)*

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Abstract

In January 2009, the IAEA EMRAS II (Environmental Modelling for Radiation Safety II) program was launched. The goal of the program is to develop, compare and test models for the assessment of radiological impacts to the public and the environment due to radionuclides being released or already existing in the environment; to help countries build and harmonize their capabilities; and to model the movement of radionuclides in the environment. Within EMRAS II, nine working groups are active; this paper will focus on the activities of Working Group 1: Reference Methodologies for Controlling Discharges of Routine Releases. Within this working group environmental transfer and dose assessment models are tested under different scenarios by participating countries and the results compared. This process allows each participating country to identify characteristics of their models that need to be refined. The goal of this working group is to identify reference methodologies for the assessment of exposures to the public due to routine discharges of radionuclides to the terrestrial and aquatic environments. Several different models are being applied to estimate the transfer of radionuclides in the environment for various scenarios. The first phase of the project involves a scenario of nuclear power reactor with a coastal location which routinely (continuously) discharges 60Co, 85Kr, 131I, and 137Cs to the atmosphere and 60Co, 137Cs, and 90Sr to the marine environment. In this scenario many of the parameters and characteristics of the representative group were given to the modellers and cannot be altered. Various models have been used by the different participants in this inter-comparison (PC-CREAM, CROM, IMPACT, CLRP POSEIDON, SYMBIOSE and others). This first scenario is to enable a comparison of the radionuclide transport and dose modelling. These scenarios will facilitate the development of reference methodologies for controlled discharges.

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METHODS AND MODEL DEVELOPMENT FOR COUPLED RELAP5/PARCS ANALYSIS OF THE ATUCHA-II NUCLEAR POWER PLANT*

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Abstract

In order to analyze the steady state and transient behavior of CNA-II, several tasks were required. Methods and models were developed in several areas. HELIOS lattice models were developed and benchmarked against WIMS/MCNP5 results generated by NA-SA. Cross-sections for the coupled RELAP5/PARCS calculation were extracted from HELIOS within the GenPMAXS framework. The validation of both HELIOS and PARCS was performed primarily by comparisons to WIMS/PUMA and MCNP for idealized models. Special methods were developed to model the control rods and boron injection systems of CNA-II. The insertion of the rods is oblique, and a special routine was added to PARCS to treat this effect. CFD results combined with specialized mapping routines were used to model the boron injection system. In all cases there was good agreement in the results which provided confidence in the neutronics methods and modeling. A coupled code benchmark between U of M and U of Pisa is ongoing and results are still preliminary. Under a LOCA transient, the best estimate behavior of the core appears to be acceptable

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