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Radiation Safety & Radioactive Waste Management

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The 2015 FNCA Workshop on Radiation Safety & Radioactive Waste Management (RS&RWM)

17-19 November, 2015, Serpong, Indonesia

The FNCA 2015 Workshop on Radiation Safety and Radioactive Waste Management (RS&RWM) was held from November 17 to 19 2015, in Serpong, Indonesia. This workshop was hosted by the Ministry of Education, Culture, Sports, Science and Technology (MEXT) of Japan and the National Nuclear Energy Agency of Indonesia (BATAN). Seventeen researchers and experts from Bangladesh, China, Indonesia, Japan, Kazakhstan, Malaysia, Mongolia, Thailand and Vietnam participated in the workshop.

On the first day, member countries gave a country report on nuclear/radiological emergency preparedness and response detailing policy, regulatory framework and legislation, emergency classes and condition, participating organization, concept of operation such as mitigation and prevention, emergency preparedness, emergency response and emergency rehabilitation and restoration, and other related activities such as human capacity building and communication and public relation.

Participants then reported progress of their consolidated report on nuclear/radiological emergency preparedness and response and agreed to complete their first compiled draft by the end of March 2016.

Figure1: The 2015 RS&RWM Workshop

An open seminar on countermeasures for nuclear/radiological emergency at nuclear facilities was held at BATAN Serpong on 18 November. The opening remarks were made by Prof Toshiso Kosako, Emeritus Professor, The University of Tokyo and Mr Suryantoro, Head, Center for Radioactive Waste Technology (CRWT), BATAN. Eight presentations were provided on each topic of countermeasures for nuclear/radiological emergency at nuclear facilities in Bangladesh, Indonesia, Kazakhstan, Malaysia, and Thailand. Japanese participants gave a presentation on overview and recent progress of FNCA, countermeasures under Fukushima Daiichi NPP accident, and human resource development programs of Japan for nuclear/radiological emergency. A panel discussion was held concerning the topics of 1) sharing information on lessons learned from nuclear/radiological accident, 2) countermeasures for external events(natural disaster etc.), and 3)human resources development.

Figure2: Open Seminar

A technical visit to the BATAN Serpong was conducted on the afternoon of 18 November 2015. Participants visited the GA. Siwabessy Research Reactor, which is a 30 MW multipurpose reactor and reached the first criticality in 1987 and the Center for Radioactive Waste Technology (CRWT), which was established as a centralized radioactive waste management facility of Indonesia in 1989. They observed evaporation system, compaction system, incineration system, and cementation system.

On the last day of the workshop, participants had a discussion on status, plans and challenges of low/Intermediate level waste disposal facilities/long term storage facilities. At the wrap up session, participants exchanged their opinions on subjects that they are interested in the field of RS&RWM and activities to be conducted in the near future.

Figure3: Center for Radioactive Waste Technology (CRWT), BATAN

Figure4: GA. Siwabessy Research Reactor, BATAN

Topics from Participating Countries

Australia

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Australian Nuclear Science & Technology Organisation (ANSTO)

Return of Intermediate Level Waste from France

The Australian HIFAR Reactor

The High Flux Australian Reactor (HIFAR) at Lucas Heights which went critical on the 26 January 1958 is the first nuclear reactor in the southern hemisphere. It was used for producing neutrons for scientific research and radioisotopes for medicine and industry. HIFAR retired in January 2007 after close to 50 years of reliable operation.

HIFAR's reactor core consisted of 25 fuel elements and has a nominal maximum thermal power output of 10 megawatts. Approximately 37 spent fuel elements (SFE) were generated by HIFAR each year. As Australia does not have a nuclear spent fuel reprocessing facility, these SFE, following a period of wet then dry storage, were sent overseas for reprocessing.

Shipment of Spent Fuel

Between 1996 and 2009, there were eight overseas shipments of used HIFAR spent fuel - four shipments to France for reprocessing, three shipments to the United States (US) as part of the US Department of Energy (DOE) Foreign Research Reactor Spent Nuclear Fuel Acceptance Program (FRR-SNF) and one shipment to the United Kingdom (UK) also for reprocessing. Each shipment was carried out safely using approved and licensed transport packages in dedicated INF-2 classified ships.

There is no waste return to Australia for spent fuel elements shipped to the USA under the US National Nuclear Safety Administration’s Global Threat Reduction Initiative (GTRI). However the inter-governmental agreements made with France and the UK stipulates the return of wastes arising from the reprocessing of ANSTO’s spent fuel back to Australia. It is internationally accepted that countries be responsible for the management of the radioactive waste they produce.

Reprocessing of Spent Fuel

Reprocessing is the chemical process whereby fissile and fertile materials in the spent fuel are recovered in order to provide fresh fuel for existing and future nuclear power plants. It offers benefits in increased uranium and plutonium utilisation, and a reduction in the volume of high level waste.

The resulting liquid waste from the reprocessing process is then melted together with glass material at up to 1300oC for incorporation into the glass matrix. The melted mixture is poured into stainless steel canisters which are welded shut after controlled cooling. The canisters undergo a wet decontamination and pin shotting to remove possible surface contamination prior to interim storage or shipment.

Wastes Arising from Reprocessing in France

A total of 1288 spent fuel elements (SFE) were sent to France in four separate shipments between 1999 and 2004. An equivalent amount of wastes from the reprocessing of these 1288 SFE was agreed upon for return to Australia.

<u>(i) Vitrified Residues</u>

A total of 20 steel canister of vitrified intermediate level waste with less than 300 W/m3 of total thermal heat were assigned to Australia. These 20 canisters were loaded into a French-designed and developed dual purpose transport and storage cask, also known as TN81. The TN81 is classified as a Type-B transport container under the international regulations for the safe transport of radioactive materials. It is designed to withstand a drop of 9 metres, temperatures above 800oC, earthquakes and jet plane crashes. In transport configuration the cask measures around 7m in length and 2.7m in diameter, and when fully loaded weighs close to 120 tonnes. Containment and shielding is provided by the thick forged steel cylindrical vessel with a welded bottom, the external lead-filled aluminium profiles and the primary lid with the associated seals.

(ii) Technological Wastes

The technological wastes incidental to the operation and maintenance of the reprocessing facility (such as contaminated gloves and tools) were also assigned pro rata to Australia. These are also intermediate level wastes, though much less radioactive than the vitrified residues.

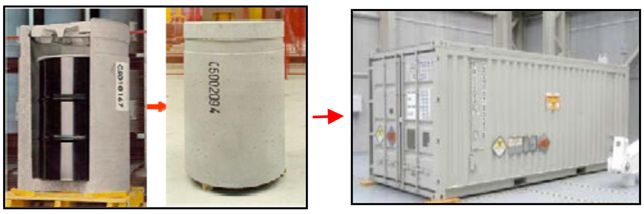

Figure1: The cemented technological wastes are cemented within overpacks and placed inside an ISO IP2 transport container, DV78.

Shipment back to Australia

Both waste packages were loaded onto an INF2 rated ship purpose-built for nuclear material shipment, at Cherbourg, France on October 2015 after a thorough inspection of the vessel by the French maritime authorities.

The position of the ship was tracked throughout its two months journey, arriving in Australian waters on December 2015. At the port, the two waste packages were unloaded from the ship and secured onto waiting trucks for delivery to ANSTO's Lucas Heights site, approximately 60km north of the port.

All maritime and road transport approvals were sought from and granted by the relevant local, national and federal regulatory bodies in France and Australia. A comprehensive security operation was also in place to ensure that safety and security were maintained throughout the operation.

Figure2: TN-81 Cask being loaded onto a Class INF2 ship

Operational Readiness Program

A comprehensive training program was set up to prepare the project team and the relevant stakeholders for the receipt of the first waste return of Australia's reprocessed spent fuel. The training format included both classroom style lectures and on-site practice with mock-up equipment. The training was conducted in phases during the course of the project, utilising subject matter experts and teaching materials sought from within Australia and overseas, and drawing upon the experience of overseas owners of similar casks.

Figure3: Hands-on training (a) Use of building crane to lift and tilt a mock-up TN81 cask from the transport frame (b) Test run of leak tightness equipment set up on a mock up secondary lid.

The training covered a broad area, including work health safety, radiation safety, regulatory, engineering, waste management, transport, security, safety and reliability. It involved staff across multiple disciplines and divisions within ANSTO, as well as external commercial partners, regulatory bodies and agencies. The training provided a forum for interaction and discussion amongst the various groups, and equipped the participants with knowledge and skills to confidently plan and execute their respective roles and responsibilities required by the project.

Arrival of Waste Packages at ANSTO

Due to the mass and size of the load, road transport of the packages from the port to ANSTO was constrained to the early morning hours and on the night of least traffic density to minimise impacts on the public. The convoy arrived at the Lucas Heights site in the early hours of Sunday morning on 6th December 2015.

Figure3: TN81 being transported on ANSTO's Lucas Heights site.

A purpose-built an interim waste storage facility, named the Interim Waste Store (IWS), was licensed by the Australian Radiation Protection and Nuclear Safety Agency (ARPANSA) in 2015 to receive and temporarily store the returned waste from France. The building is equipped with a 140T DG rated crane.

The DV78 with the technological waste was delivered first to the IWS and placed into position using the building crane. The TN81 was driven into the building next on the back of a 7-axle trailer.

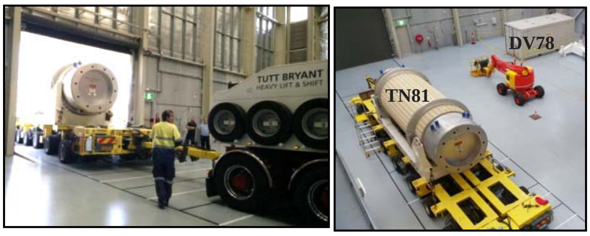

Figure4: (a) TN81 arriving into the IWS on the back of the transport trailer. (b) DV 78 and TN81 inside the IWS.

A thorough radiation dose and contamination survey was conducted on both waste containers by the radiation protection advisors and health physics surveyors. No contamination was found. The dose rates on the TN81 and DV78 packages measure 0.011mSv/h and 0.350mSv/h respectively, well below the maximum permitted transport container surface dose rate of 2 mSv/h according to the IAEA transport regulations.

Setting up of TN81 for Interim Storage

In order to store the TN81 in the facility, a series of operations were undertaken to convert the cask from its transport configuration into a storage configuration over the following few days.

Day	Activities
1	Remove impact limiters. Tilt and lift from trailer. Locate within scaffold.
2	Prepare secondary lid. Mount leak tightness monitoring equipment. Evacuate inter-lids space.
3	Conduct leak tightness test. Install anti-aircraft crash cover.
4	Connect to SCADA system for continuous monitoring and recording. Remove scaffold.

Table 1: Program to set the TN81 into storage configuration.

This included firstly the removal of the front and back impact limiters. A two-legged lifting beam was then attached to the front trunnions of the cask. Using the

140T DGR building crane, the cask was carefully tilted and lifted off the transport frame on the back of the trailer. The cask is then located in its storage position and a scaffold built around it to allow access to the top of the cask.

Figure5: (a) Tilting and lifting of the TN81 from the transport frame. (b) Handling team posing in front of the vertically positioned TN81.

On the second day, the secondary lid was cleaned and fitted with two seals – a metallic seal and an elastomer seal. The monitoring system consisting of a pressure vessel and three pressure transducers was also secured onto the secondary lid before bolting the prepared lid onto the cask over the primary lid.

Figure6: (a) Lowering of the secondary lid over the primary lid of the TN81. (b) Positioning the bolts for securing the secondary lid.

After securing the secondary lid, the leak tightness equipment was set up and the inter-lid spaces evacuated overnight.

On the third day, the inter-lid space was systematically pressurised to 6.5 bar absolute, first with nitrogen, then with helium. This was followed by a leak tightness test to ensure the required seal was achieved and the minimum leak rate was not exceeded.

On the fourth day, the TN81 was connected to the monitoring and logging system within the IWS for continuous monitoring and recording of the pressure signals from within the inter-lids space of the TN81 and the temperature of the IWS. The SCADA system is linked to the site control centre for notification of excursions from any alarm set points.

Figure7: TN81 in storage configuration inside the IWS.

This project completes a significant milestone in the management of ANSTO’s backend fuel cycle. It has involved not only multi-disciplinary participation and dedication across ANSTO, but also close collaboration and cooperation with commercial partners, industry experts, regulatory bodies and governmental bodies both nationally and internationally.

National Waste Repository

The Australian National Radioactive Waste Management Facility (NRWMF) is anticipated to be operational by the end of the decade, at which time the French returned waste will be transported from the IWS in ANSTO for long-term storage in the NRWMF.

Bangladesh

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Bangladesh Atomic Energy Commission (BAEC)

Gamma Analysis of Solid Radioactive Waste Produced in the Nuclear Facility

Introduction:

The decommissioning and dismantling of nuclear facility after their service life involves the necessary disassembling, handling and disposing of a large amount of radioactive equipment and structures. To reduce this undesirable volume to the minimum and to successfully plan the dismantling and disposal of radioactive materials to storage facilities, the activations of the structures should be accurately evaluated. To optimize the mass flows going to waste disposal it is important to determine the radionuclide present in the waste materials generated due to the long-term induced activity. In particular, the graphite material that has been used in the reactor environment for several years requiring an appropriate disposal, hence minimise the total cost of dismantling and decommissioning.

Accurate characterisation requires the representative samples be taken from material to be chracterised. The spectrum of the radiation from the sample is measured and from this the constituents and their activities are determined. Assuming that the sample is representative of the entire component, the total activity of the material concerned per unit weight can be deduced.

The graphite plugs which were used in the vertical beam tube and irradiated for several years in the reactor environment categorize as solid radioactive wastes, these wastes are collected in polythine bags and safely stored in the waste storage facility. The activated graphite sample collected from research reactor facility was analysed for strong gamma lines by using HPGe detector coupled with Genie software. A series of activity measurements with a few representative waste containers were performed. Before the experiment took place a written system of work was produced.

Experimental Methods

To aid risk assessment a non-active run was initially performed, with a sample for an unirradiated graphite sample. Contamination was checked before scraping. To avoid finger cuts from sharp blades, latex gloves and leather gloves were worn. In order to analyse the graphite plug three representative samples were taken from activated graphite. By using β/γ survey meter the maximum average dose rate and surface activity of the sample was recorded 2.80 μSv/h and 8.48 Bq/cm^2 respectively. A scraping technique was performed to take/collect the sample from the active graphite. Polythene, latex and leather gloves were worn to avoid cuts and contamination safety spectacles were used to protect eyes from dust during the scraping of the graphite samples. Protective clothing, lab coats and masks and pocket dosimeter, TLD were worn before entering the waste storage facility in order to collect the graphite sample.

Any loose particles from the graphite were picked up on plastic tape. The plastic tape was put into a 250 ml plastic pot. The sample was then collected in the plastic containers. The sample was then put into a lead pot to take up to the counting room.

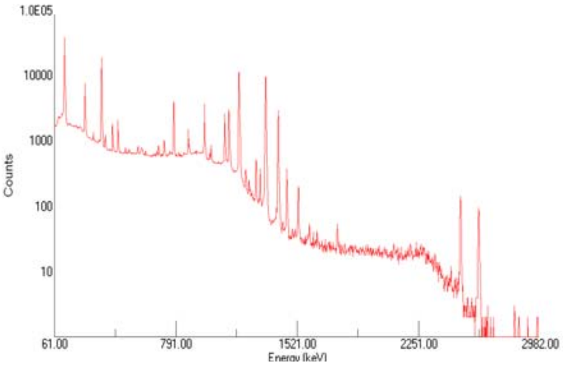

Figure: Activated graphite material & corresponding γ -spectrum obtained from HPGe detector

Each graphite sample has been weighed and measured on a calibrated HPGe detector with 40% relative efficiency coupled with Genie-2000 software for γ activity. In order to calculate the activity of the graphite sample, the energy and efficiency calibration was done using a 152Eu standard source and the spectrometer was set to detect the dominant gamma energy. The standard was placed on the top of detector within the shielding arrangement in the counting room of HPRWMU. The counting time of the graphite sample was 10,000 sec. The sample was placed in the same position as the standard source for efficiency calibration. A same geometry was chosen for reference standard source for efficiency calibration. An unirradiated graphite sample was also counted for 10.000 sec.

Results

The activated graphite samples have been measured experimentally for gamma activity using HPGe detector coupled with Genie-2000 software. The dominant gamma lines found in the spectrum are: 1173.2 kev (0.99), 1332.5 kev (0.99), 121.8 kev (0.284), 344.3 kev (0.266), 1408.02 kev (0.21) and 723.3 kev (0.20), 1274.5 kev (0.355). The identified respective radionuclides in the activated graphite samples are 60Co, 152Eu and 154Eu respectively. From the strong gamma lines the calculated average specific activity concentrations of the measured radionuclides are: 6.3 Bq/g, 15.2 Bq/g, and 0.62 Bq/g for 60Co, 152Eu and 154Eu respectively. As expected there is no measurable gamma activity found for the unirradiated graphite sample. The result obtained in this work could be used to estimate radioactive waste generated during decommissioning activities and to determine the appropriate methods for dismantling and demolition for this particular nuclear facility.

Indonesia

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Integrity Test of the SNF Using Sipping Test Method

INTRODUCTION

In the national policy an strategy on radioactive waste management, SNF is classified as HLW (Goverment Regulation No. 61 Year 2013 Radioactive Waste Management). Indonesia follows open nuclear fuel cycle that the SNF will not be recycled. CRWT as the centralized radioactive waste management has responsibility to manage the SNF from G. A. Siwabessy Multi Purpose Reactor 30 MW.

CRWT has Transfer Channel Interim Storage for Spent Fuel Installation (TC-ISSF). The facility was built in 1993 and designed by AEA Technology, United Kingdom (UK-AEA). The main function of TC-ISSF is to receive and store the SNF and other irradiated materials. Main part of the TC-ISSF is water pond with dimension of 14 m in length, 5 m width and depth of water 6.5 m. At the present there are remaining of 245 elements of SNF on the TC-ISFSF. The ISFSF has the maximum storage capacity of 1448 elements of SNF. Fuel element used in GAS_MPR is Material Testing Reactor (MTR) fuel type with U-235 enrichment of 19.75%, dispersed in Al matrix form plate. Control fuel element is part of the reactor control rod that used to control neutron flux. Based on GAS-MPR technical spesification the fuel element must be removed from reactor core after reached 56% of burn up.

One of the requirements on radiation safety and operation of a research reactor is the absence of radionuclide release from fission products to the environment. Sipping test is one of non-destructive testing technique for detection of the cladding by detection and identification the presence of fission products in the water such as Cs-137 and others.

1. THEORY

SNF is a fuel element type Material Testing Reactor (MTR) with a cladding material is AlMg2. Being used in the core of the reactor fuel elements produce various fission products. Fission products trapped in the matrix of fuel elements can also separate out of the fuel matrix element. Noble gas group is easy to separate such as Xe-138, Kr-87, Kr-88, Kr-85m and Xe-135. The others fission products and transuranics may leak to the environmet since the isolation of the cladding has low integrity such as defect, pitting or crack. Some of fission products like Cs-137 pass through the cladding by diffusion mechanism. The sipping test is appropriarte method to determine the integrity of SNF.

2. METHOD OF SIPPING TEST

The sipping test procedure:

1. SNF is transferred from its rack and placed into the basket of test tube that located 2 m under water surface.

Figure1: Transferring spent fuel from rack to sipping test basket

2. The sipping tube is raised by rolling up the wire rope sling with an electric motor until the top of the test tube appears slightly above the water surface.

Figure2: Raising sipping test tube

3. Demineralised water is pumped into the test tube through water inlet port at the bottom of the tube. It is estimated that 5 times test tube volume of demineralised water is needed to ensure that water inside the tube has been fully replaced by the demineralised water. Capacity of the test tube is 0.053 m3, so five times of it is 0.265 m3. It takes about 15.9 minutes to pump demineralised water using a pump with capacity of 1 m3/hour.

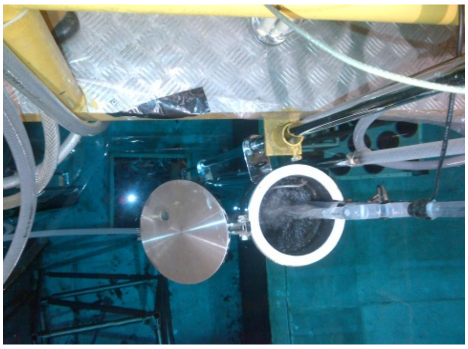


Figure3: Filling sipping test tube with demineralised water

4. Water sample is taken from the test tube for radioactivity background analysis.

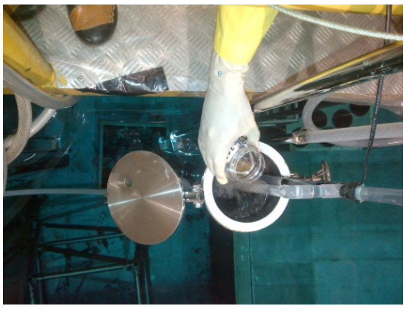


Figure4: Initial water sampling

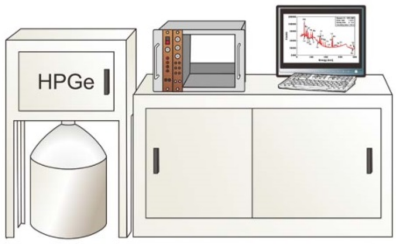
5. Let the test tube with spent fuel inside for 4 hours.
6. Air is blown into the test tube to stir the water to be agitated.

Figure5: Stirring water inside sipping test tube

7. Water sample is taken from the test tube for radioactivity analysis.

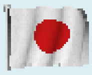
Figure6: Water Sampling

8. Radioactivity analysis is conducted to determine the activity concentration of Cs-137.
9. Previous step 5 to 8 are repeated with an interval of 4 hours.
10. Graph of Cs-137 activity concentration versus time is created using the analysis data obtained. Based on the graph, it can be predicted whether the spent fuel has leaked or not.
11. Water footage taken via pipeline output and put into a test tube of plastic bottles with a volume of 500 ml.
12. The activity of noble gases was identified and measured using gamma spectrometry device.



3. RESULT

The sipping test method is one of the techniques to find out possible leakage on the fuel claddings, by detecting the presence of fission product nuclides in fuel elements soaking water. It has been identified that 4 spent fuel element releasing fission products with maximum activity Cs 137 : 4,00 x 10^{-7} µCi/mL. From these results indicate that there are no leaks in the spent fuel detected.



Japan

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A Measurement Quality Assurance Activity for Radiation Protection in Japan: Role of the Institute of Radiation Measurements

1. Measurement Quality Assurance of Radiation Protection Instruments in Japan

Radiation measurements are very important and necessary in and around any types of nuclear and radiation facilities. In view of radiation protection purposes at these nuclear related facilities, reliability of the radiation measurement results is of highly great interest. Especially after the accident at TEPCO Fukushima daiichi NPP(Nuclear Power Plant), public interest has become higher than before and various types of radiation protection instruments have been commonly used for the measurement of radiation even in the general environment. However, reliability of radiation measuring instruments (dose and dose rate meters, radioactivity measuring apparatuses) is not always sufficient. To get reliable measurement results, maintenance and calibration system for keeping the performance of these instruments in good level (namely "measurement quality assurance") is strongly required.

Calibration and regular maintenance for the instruments need 1) reference standards with traceability to primary standard and 2) engineers having good capacities. Additionally, national and/or international quality assurance programs are required to get higher reliability of public and radiation workers.

In Japan, AIST (National Institutes of Advanced Industrial Science and Technology) keeps primary standard on radiation dose and dose rate quantities, and some organizations have second or reference level standards of these quantities with traceability to primary standard. Calibration system and methods for radiation protection instruments is standardized as a Japanese Industrial Standard (JIS) object (JIS Z 4511[*1)]; Methods of calibration for exposure meters, air kerma meters, air absorbed dose meters and dose-equivalent meters). Traceability system in Japan is shown in Fig.1. Most of calibration laboratories conduct calibration of radiation protection instruments according to the method of JIS Z 4511 with standards traceable to primary standard.

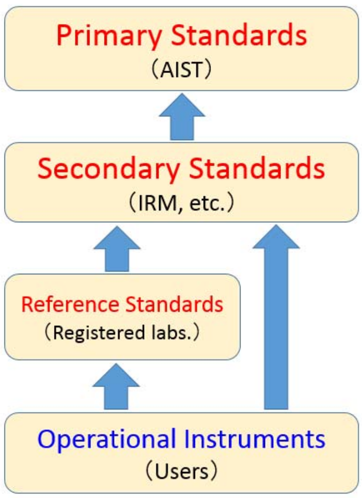

Figure 1: Traceability System on Radiation Dose Quantities in Japan

In Japan, more officially-certified system called Japan Calibration Service System (JCSS) for calibration laboratories (they satisfy ISO/IEC 17025[*2)] requirements) has been also applied to get public-reliability on measurement results for various quantities. In the field of radiation measurements, eight organizations are registered as JCSS-fulfilled calibration service laboratories, and three of them have also been registered as an internationally accredited calibration service supplier (sufficient for ILAC-MRA system; International Laboratory Accreditation Cooperation – Mutual Recognition Arrangement). As one of three MRA registered calibration service suppliers on radiation measurements in Japan, activities of the Institute of Radiation Measurement are introduced in the following.

2. Activities of Institute of Radiation Measurements*3)

In Japan, development of nuclear-related technologies started in 1956, and since then research and development activities on radiation measurements had also been mainly conducted at JAERI (Japan Atomic Energy Research Institute; One of former organizations of Japan Atomic Energy Agency: JAEA). In 1980, JAERI constructed a large scale facility for R&D on radiation measurement for radiation protection purposes, named Facility of Radiation Standards; FRS, and intended to disperse the productions of R&D activities on radiation protection measurements. The Institute of Radiation Measurements (IRM) was established in the same year (1980) to disseminate the fruitful productions of R&D activities of JAERI and to build up reliability of radiation protection measurement results throughout social communities in Japan. IRM is now contributing to society through supplying reliable calibration services of various types of radiation protection instruments and conducting relevant activities.

The business of IRM categorizes into 4 different activities, R&D activities on radiation measurements, calibration services for radiation protection instruments, measurements on radioactive materials (including survey of working environments) and trainings of young generation on radiation measurement techniques. Additionally, many other activities to spread basic radiation knowledge throughout public, such as seminars on radiation, etc., are also conducted.

Some special activities of IRM are described as follows;

1) Recently, needs for education and trainings on radiation knowledge are growing for very young generations such as high school students and thus IRM is now strongly supporting to conduct those programs by training high school teachers.
2) Since 2009, IRM has planned an information exchange meeting for experts on current radiation measurement technologies and carried out it almost every year. After TEPCO NPP accident in Fukushima prefecture, various types of radiation and/or radioactivity measurement instruments have been used at many places and then the themes of the expert meeting are changed into such related subjects as measurement methods at emergency situations, etc. Most recently, IRM made an expert meeting on agricultural food products monitoring methods for contamination area affected by Fukushima daiichi NPP accident. (See Photo 1)

Photo1: Expert Meeting on Radiation Measurement Technologies (January, 2016 Tokyo)

3) IRM has been conducting an investigation measurement program for widely contaminated area in and around Fukushima prefecture since FY2013. This investigation measurement (called In-situ measurement) is made with a handy-type Ge detector and is useful for analysis of distribution of radioactive materials after the NPP accident. Photo 2 shows an In-situ measurement work.

Photo2: In-situ Measurement in environment (Fukushima Prefecture)

4) In Japan, there are 4 different individual monitoring service companies that cover approximately 5 hundred thousand radiation workers. In order to verify that the uncertainty of each service will be within the permissible level, IRM is playing a central role of individual dosimetry inter-comparison program; blind irradiation and report of the measurement evaluation results to each companies.

IRM is a small organization (about 50 personnel) but, as introduced above, is now playing an important role for securing the reliability (quality assurance) on radiation measurements activities in Japan.

References:

1) JIS Z 4511; Methods of calibration for exposure meters, air kerma meters, air absorbed dose meters and dose-equivalent meters, Japan Standards Association (2005) (In Japanese)

2) ISO/IEC 17025; General requirements for the competence of testing and calibration laboratories (2005)

3) Brochure of Institute of Radiation Measurements (2013) (In Japanese)

Malaysia

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Borehole Disposal for Disused Sealed Radioactive Sources: Malaysian Initiative

Introduction

Sealed radioactive sources are widely used by many industries, universities, hospitals and research institutes in Malaysia. All these practices have led to the generation of radioactive waste, i.e. disused sealed radioactive sources (DSRS), which some are intensely radioactive. These DSRS have various types of radionuclides which all need to be management and disposed of safely although some of them can be recycled and reused. In Malaysia it is mandatory by law to repatriate these DSRS to the country of origin. However, sometime this cannot happen due to some reasons i.e. the manufacturer or suppliers close their business or bankrupt. If these DSRS are not managed properly, they can represent a significant hazard to human health and the environment. Therefore the DSRS need to be sent to Malaysian Nuclear Agency as National Radioactive Waste Management Centre for safe keeping after getting approval from the regulatory body.

Figure1: Disused sealed radioactive sources

Storage in a secure facility can be considered as an adequate final management option for sources containing quantities of short-lived radionuclides, which decay to harmless levels and the radioactivity below than regulatory control within a few years. However, for most other sources a suitable disposal option is required. Near-surface disposal facility is not suitable for the source constitutes a high, localised concentration. Deep geological disposal offers the highest level of isolation available within disposal concepts but such facilities are under consideration for the disposal of spent nuclear fuel, high level waste and intermediate level waste in a number of countries. However, they are expensive to develop and only viable for countries with extensive nuclear power programmes. Therefore increasing attention has been given to the disposal of DSRS with long lived radionuclides and very high activity sources in borehole facilities with a view to providing a safe and cost effective disposal option for limited amounts.

It is the right time for Malaysia to consider the borehole disposal as the ultimate solution to the increasing number of DSRS stored in the current facility. This is because:

- Many DSRS cannot be returned to the manufacturer.
- Inadequate storage of DSRS presents
- Continued storage perpetuates the risk and places burden on future generations
- Concerns over safety and security.
- High chances of accidents to occur.

Borehole Disposal Concept

The Borehole Disposal Concept (BDC) has two main safety functions – it allows waste packages to be placed deep below the ground where humans cannot interfere with them, and it provides access to an environment that is stable and capable of keeping the waste packages intact for many thousands of years. Physically, a BDF is about a quarter of a metre diameter and more than 30metres to 300 metres deep. It will be drilled and constructed using standard techniques developed for deep water abstraction, oil exploration etc.

The borehole will usually be lined with mild steel or high density polyethylene tubing with concrete pumped into the gap between the lining and the host rock. The purpose of the lining is to aid the package emplacement process, especially by supporting the borehole wall (see Figure 2). At the bottom of the borehole is a concrete plug. Once the concrete has set, a fresh batch of concrete is placed in the bottom of the hole and the first disposal package is lowered into this so that it sinks through the fresh concrete and stands on the concrete plug. When this batch of concrete is set, the top of the concrete forms the platform for the next disposal package. The process is repeated until the required number of packages has been emplaced. The topmost disposal package will always be at least 30 metres below ground level.

With all the packages in place, the section of borehole lining above the topmost package is removed. A steel deflection plate is placed above the topmost package to help prevent inadvertent drilling into the disposal zone and the (now unlined) top of the borehole is filled with concrete to within 2 metres of the surface. The final few metres are filled with native soil. This will make the borehole difficult to locate without specialised equipment. Figure 1 shows the schematic layout of a borehole disposal facility.

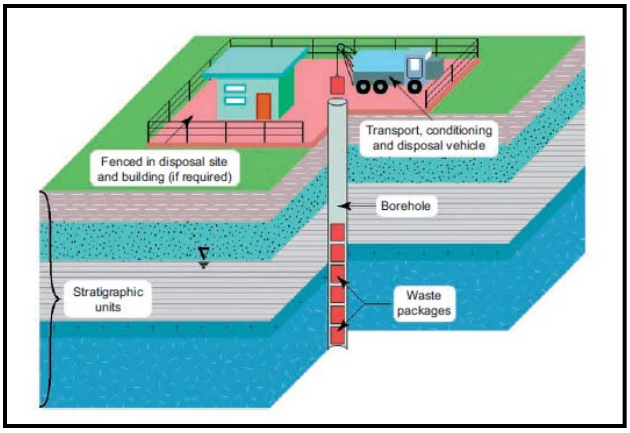

Figure2: Schematic layout of a borehole disposal concept (Ref 1)

Why is Malaysia interested with borehole disposal concept?

Malaysia is interested to develop borehole disposal concept. A study to know the suitability of borehole disposal concept in Malaysia has been done. It was found that the advantages of the borehole disposal as the following:

- Adjustable depth – the depth can be adjusted between 30 metres to 300 metres depend on the number of disposal canister to be disposed of;
- Safety – comply with the "defence in depth" concept where there are 8 tiers defence which include radioactive double encapsulated stainless steel, stainless steel 316L capsule, concrete barrier, stainless steel 316L disposal canister, concrete backfill, steel or HDPE casing, concrete disturbed zone backfill and geosphere/host rock.
- Security - robust to human intrusion
- Small footprint
- Simple, easily available technology
- Small volume excavation

At present a Malaysian Nuclear Agency is carrying out a feasibility study to develop BDF. Desk study on geology, hydrology (surface and underground), geochemistry, geomorphology, meteorology and seismic has been done. The feasibility study also includes studying several potential sites and determining whether the potential site is suitable for accommodating a BDC. The suitable site will only be considered for future work if provided the safety assessment demonstrates of its feasibility, otherwise the facility will be sited at a yet to be determined alternative location.

DSRS Inventory, Characterisation and Conditioning

Currently, there are only 4 DSRS in Category 1 and Category 2, and more than 12,000 DSRS are in Category 3 to 5 in Malaysia. The current priority is to dispose of the Category 3 to 5 DSRS under the borehole disposal concept. With regard to the Category 3 to 5 DSRS, the sources will be identified from the current inventory, characterized and verified (Figures 3 and 4). The sources are then dismantled from it shielding container and emplaced into a stainless steel capsule (See Figure 5). The capsule will be remotely welded, leak tested and placed into a shielded concrete-filled 200-L steel drum. The drum with welded capsule will be transferred to a storage facility. Emplacement into a disposal container and welding will be done in the next stage of the process. All these work will be done within a properly shielded and controlled work area (See Figure 6).

Figure3: Sources Characterization

Figure4: Sources Characterization

Figure5: Conditioning of DSRS done by Nuclear Malaysia Staff

Figure6: Work Area for Conditioning of DSRS

This project is expected to develop a Borehole Disposal Facility for disused sealed radioactive sources. The current and future generation will benefit from this project. The success of the project would set as an example and experience can be shared with other countries as borehole disposal is not available anywhere else in the world.

Reference

1. International Atomic Energy Agency, Borehole Disposal Facilities for Radioactive Waste, IAEA Safety Standards Series No. SSG-1, IAEA Vienna (2009)

Thailand

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Safety and Security during Transport of Radioactive Materials and Radioactive Waste

The Regional Workshop on Safety and Security during Transportation of Radioactive Material was conducted in Bangkok, hosted by International Atomic Energy Agency (IAEA) and Office of Atoms for Peace (OAP), during 22-26 June 2015. The purpose of the workshop is to present and discuss experiences and best practices in the development and implementation of safety requirements and security recommendations on the transport of radioactive material, national and international cooperation and prevention of delay/denial of shipment of radioactive material.

The expected outputs of the workshop are:

1. clear understanding of the objectives of the IAEA safety requirements and security recommendations for transport of radioactive material;
2. general understanding and appreciation of the similarities and differences between safety requirements and security recommendations during transport of radioactive material;

3. general appreciation of the interface between transport safety and security and potential impacts of this interface on the efforts to strengthen the compliance assurance regime for the safe transport of radioactive material
4. presentation of experiences and good practice in the development and implementation of recommendations, national and international cooperation and prevention of delay/denial of shipment of radioactive material transport

There were 25 participants from the Member States and Entities in Asia and the Pacific, participating in the regional TC project RAS/9/067, namely: Afghanistan, Australia, Bahrain, Bangladesh, Cambodia, China, Fiji, Indonesia, Islamic Republic of Iran, Iraq, Israel, Jordan, Kuwait, Lao People's Democratic Republic, Lebanon, Malaysia, Marshall Islands, Mongolia, Myanmar, Nepal, Oman, Pakistan, Palau, Philippines, Qatar, Saudi Arabia, Sri Lanka, Syrian Arab Republic, Territories under the Jurisdiction of the Palestinian Authority, Thailand, Vietnam and Yemen, as shown in Figure 1.

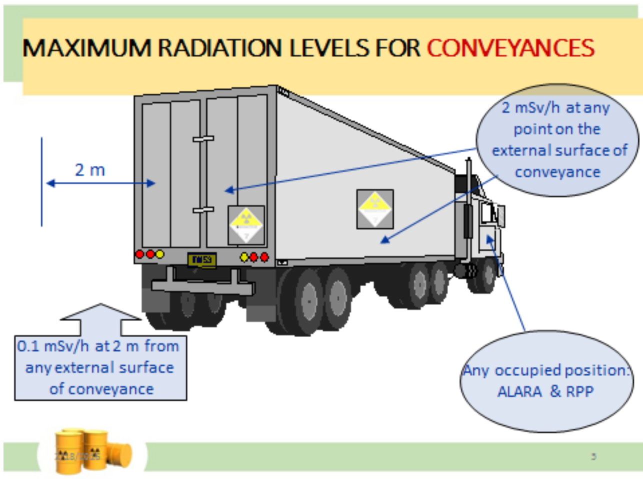
Figure1: Participants from Member States

The workshop consists of presentations, discussions, and practical sessions on safety and security during transport of radioactive material including national and international cooperation and prevention of delay/denial of shipment.

The technical visit was conducted to visit a company which carried on an activity of transportation of radioactive sources for gamma radiography , as shown in Figure 2.

Figure2: Transportation of radioactive sources for gamma radiography

Radiation Safety during Transportation



CONTAMINATION LIMITS

The overriding requirement concerning the presence of radioactive contamination is ALARA

Contamination limits are established for:

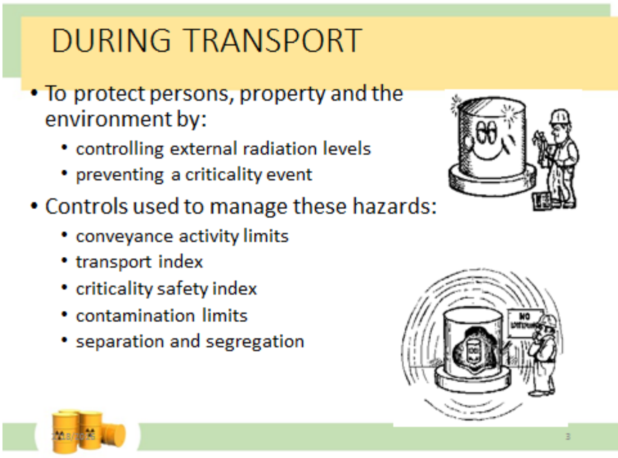
-External surfaces of all packages

-Internal surfaces of tanks, over-packs, IBCs and freight containers

Non-fixed contamination on the external surfaces of packages and on the internal and external surfaces of overpacks, freight containers, tanks and intermediate bulk containers shall be kept as low as practicable and shall not exceed the following limits --

For β, ϒ and low toxicity α emitters: 4 Bq/cm2

For all other α emitters: 0.4 Bq/cm2



Transportation of Radioactive Materials and Radioactive Waste in Thailand

Thailand Institute of Nuclear Technology (TINT) has the transportation service for radio-isotopes for medicine and radioactive waste in Thailand as shown in Figure 3.

Figure3: Transportation of Radioactive Materials and Radioactive Waste in Thailand

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The FNCA Framework