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Hazard and Operability Study for Design of Near Surface Disposal in Serpong Site

Introduction

In order to increase capacity building in nuclear technology, especially in radioactive waste management, BATAN has a near surface disposal design for the Serpong site. It is designed to accommodate low level radioactive waste currently stored in the interim storage at the Center for Radioactive Waste Technology - BATAN.

The very basic principle of disposal is that disposal facilities are placed, designed, constructed, operated, closed and decommissioned in such a way that their workers, publics, and the environment are protected from radiological hazards. In preparing the design of Near Surface Disposal (NSD), the dimensions of waste processed and the dimensions of building its own NSD facilities have to be prepared and become the basic data that must be calculated. The size, weight, and amount of waste to be placed in the NSD should be estimated, it is necessary to maintain the condition of the packaging and the placement of the waste.

Near Surface Disposal building foundations should be able to support the overall load. The disposal site to be built is not only used as a mere waste disposal, but needs to be equipped for various things, such as the addition of buildings to complete the disposal operation. The layout and addition of buildings vary from one disposal facility to other disposal facilities, depending on need. Most importantly, the supporting facilities within the disposal area should be designed in such a way as to facilitate the decontamination of an accident resulting in contamination within the facility's disposal area. Roads connecting waste receiving areas and other supporting areas with disposal areas should be designed separately so that disposal activity traffic does not affect disposal structures that have been filled with radioactive waste, and the presence of supporting facilities does not interfere with disposal activity traffic.

Geomorphologically, Serpong site relatively meets the requirements with hilly and non-sloped contours, the slope is small and the intensity of the geomorphology process is also small [4]. Although in this aspect the existing site has met the criteria, but optimization is still done by placing disposal facilities at locations as far as possible with rivers, slopes, and zones that have large erosion potential.

The construction of a demo disposal facility is planned for an area of 1197.16 m2 with a size of 34.6 m x 34.6 m. Engineered vault shape with reinforced concrete vault walls with 0.6 m wall thickness. Vault is built with a depth of 2 m and the foundation of the building is located 4 m above the highest groundwater level. Vault is divided into 2 (twin) compartments separated with 0.6 m thick reinforced concrete walls. The volume of each vault space is 594 m3 with a length of 18 m, width of 6 m and a depth of 5.5 m. Right compartment for disposal of waste package in 200 liters drum with high drum dimension 85 cm and diameter 60 cm with weight ± 0,486 ton. While the left compartment for the waste package in a 950 liter concrete shell with dimensions of height 1.4 m and a diameter of 1.3 m with a weight of ± 6.4 tons. For the left compartment, 200 liters drums are arranged in 15-lane, 15 row, and 3 stack modules, so that in one module it can hold 675 200-liter drums. For the left compartment the 950 liter concrete shell will be arranged into 6 lanes, 6 rows, and 2 stacks, so that in one module can accommodate 72 concrete shells.

Methodology

This study is an analytical descriptive study to identify hazards on the design for NSD. The analysis is carried out on any potential failures of the disposal system and its supporters, and provides descriptive descriptions of the factors leading to such failures and their impact on safety, working areas, the environment surrounding NSD facilities, and financial losses. In this study, an analysis of the design of disposal system components consisting of the disposal system itself (cover, vault, drainage, waste package, backfill), site characteristics, and access control of disposal facilities.

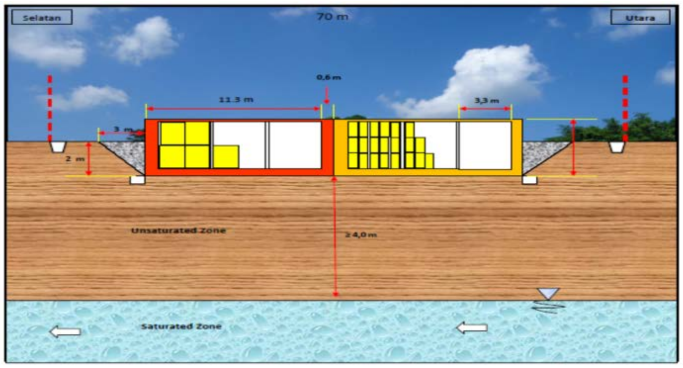
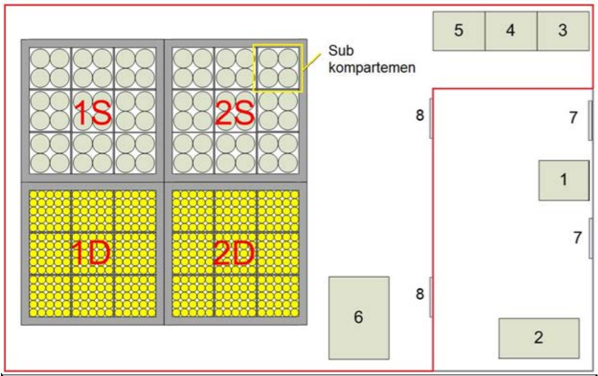


Figure 1. Cross-Sectional View for Design of NSD



1S, 2S : Concrete Shell Storage	5 : Maintenance Building
1D, 2D : Drum Shell Storage	6 : Interim Storage
3 : Laboratorium	7 : Main Gate (In – Out)
4 : Utilities Building	8 : Restricted Area Border

Figure 2. Layout for Design of NSD

HAZOPS is chosen because that it is a systematic, thorough, and complete method, and can be used in the design stage of an installation or production process. The design stage is the right time to do the HAZOPS study. If the results of the HAZOPS study recommend corrective action against the design, then it will not cost more than later when the facility or process has been running.

Results and Discussion

The study node of the radiation safety risk analysis is the barriers in the disposal system. Barriers are nodes of study because they are the determinants of the reliability of the disposal system to support the radioactive waste stored therein, either as shielding for radiation exposure or containment of the possible release of radionuclides to the environment.

Generally, the barriers are divided into 4 (four) disposal components i.e. disposal engineering system, waste package, site, and access control. The disposal system consists of cover, vault (including bottom cover), drainage, backfill material and buffer used. As for the waste package, the integrity of the waste matrix is the determinant of the reliability of the waste package as a supporter. For site risk assessment, various site-related risks will be assessed including geomorphology, lithostratigraphy, geological structure, engineering geology, hydrogeology, and potential natural disasters. Whereas access control has a potential failure of access should not, whether it is access without intrusion, or access with intrusion. The parameters present at this study node are determined at the design stage, so the results can to be a recommendation for the design.

Tabel 1. Study Node for Design of Near Surface Disposal

No.	Study Node	Parameter
1.	Cover	Permeability
2.	Backfill	Permeability
3.	Drainage	Flow Rate
4.	Waste Matrix	Integrity (compressive strength, density, leachate test)
5.	Vault (include bottom cover)	Radiation Dose Rate, Permeability
6.	Access Control	Incongruity
7.	Site (Natural Barrier)	Permeability, distance of facility to surface water body, soil bearing capacity

1. Cover

Cover is built after the operation phase of the disposal facility. Cover must satisfy 3 (three) design criteria, namely impermeability, reliability, and protection. In the impermeability criterion, the amount of rainwater that enters the cover and reaches the waste package must be very low to avoid leaching and radionuclide migration. The criterion of reliability is that the impermeability of the cover must be maintained until the period of institutional control or waste ends for approximately 300 years. The material to be used should be selected to ensure its ability over a long period of time. While the criteria of protection require the cover to be resistant to external forces such as erosion, chemical processes, and organisms. Of all these criteria, permeability becomes a parameter that determines the reliability of the cover. For that cover must be made thick enough and made of material that resistant from external force. If the cover fails to maintain its performance, significant water infiltration will occur and will impact on the performance of the waste packaging that will be the direct cause of the release of radionuclides from the waste packaging.

Serpong Nuclear Area is an area that has high rainfall and tropical climate so that proper cover design should be considered for its function as a drain and rain water drainer can run well so that rain water does not contact with radioactive waste package. The recommended cover design is as follows:

- The cover component is a multi-layered cover, consisting of an uptake layer that also function as a biological protective layer of vegetation with soil, rocks, and sand, then a layer of low permeability that can be clay or bentonite compacted, primary and secondary with membrane layers placed between 2 (two) drainage layers;

- The thickness of each layer depends on the type of radioactive waste. Since the location of the potential disposal facility has high rainfall, the thickness of the drainage layer should be thick enough with the dome design having a slope of 2o – 3o;
- The drainage system is a part of the cover design comprising primary and secondary layers so it must be carefully calculated in the overall design to ensure the function of radioactive waste insulation.

The recommended cover design is the result of cover performance analysis done by using El-Cabril cover design.

2. Backfill

Backfill or filler material is engineered barrier added to sub compartment after fully charged. Backfill fills the gap between the waste package and the layer between the waste package and the layer above it. The main function of the backfill is as a high permeability layer that passes water to immediately descend to the bottom of the disposal if there is water infiltration (either surface water or ground water) into the vault thus reducing water contact with packaging and waste matrix. If the backfill fails to perform its function, then the chance of water contact with the waste package will be greater, it means the chance of damage to the waste package will also be greater.

Anticipation of potential failure of backfill material can be done by laboratory scale experiments on the type of material that has high permeability with the existing site environment. According to the results of the study, a mixture of bentonite and quartz sand (more quartz sand composition) was used.

3. Drainage System

The drainage system in the NSD facility serves to drain the infiltrated water into the facility to the control pool so as not to accumulate in the facility. If water accumulates in the facility, it will increase the moisture inside the facility that can enlarge the waste pack degradation factor. And the worst possibility can cause flooding in the facility that inundated the waste package. Risk control measures that can be done to prevent the failure of this drainage system is by engineering drainage-shaped gutter with a slope of 1%.

4. Waste Matrix

As the waste packet node representation, waste matrix is used which can be measured its characteristics with its own integrity. Radioactive waste that has been treated with a mixture of cement and sand, both in 200 liters drums and 950 liters of concrete shell, always tests quality to determine the integrity of the waste matrix. The integrity of the waste matrix can be seen from the results of compressive strength test, density measurement, and leachate test. The results of the compressive strength test and leachate test show how strong the waste matrix from the risk of breakage and how resistant to the waste matrix will not decompose which may lead to the release of radionuclides from the waste package. As a recommendation, it is advisable that the waste package to be stored in the NSD facility is the waste treatment package for the last 10 years. This is to anticipate the degradation of waste packages that have occurred during temporary storage. If not possible, then the package of waste to be stored viewed the condition of packaging, if it has experienced degradation should be over packaging.

5. Vault (include Bottom Cover)

Vault function as the primary supporting of the disposal system. In addition it function as a major supporter of the disposal system, i.e. as a radiation barrier from radioactive waste and as a barrier (vault and bottom cover) to the release of radionuclides from a packet of radioactive waste to the environment. If this node fails, it will be indicated by 2 (two) major impacts, namely the increase of radiation exposure and the release of radionuclides. For a radiation exposure barrier/ shielding, the selection of materials that have sufficient half-value layers to withstand the radiation from the encoded radiation source. While its function as a containment is highly dependent on the quality and homogeneity of its constituent material in order to survive the hydrological/ hydrogeological and climatological environment around the NSD facility tread.

As an effort to control the risk of exposure to radiation outside Vault, in the pre-design stage it is necessary to know in advance the inventory of radioactive waste to be stored in the NSD facility, be it radionuclide, activity, radiation exposure, and amount of radioactive waste to be stored in the facility. If this inventory is known, then the design for the vault can be determined, such as material with sufficient half-thickness as well as the thickness to be satisfied. For the recommended vault design is K350 reinforced concrete 60 cm thick. This concrete other than as a radiation shielding, also serves as a reinforcing structure in order to optimize the disposal placement in the Serpong Nuclear Area site. As an early warning effort, routine monitoring of the pool can be monitored to anticipate releases of radionuclides in order not to release to the environment. As a supporter of radionuclide release, and taking into account the site conditions, rainfall, relatively shallow groundwater surface and the potential for movement of the soil, a below grade vault design is recommended.

6. Control Access

The main function of access control is to restrict unauthorized parties from entering the NSD facility. If this access control fails, there may be intrusions that cause direct or indirect disposal facility damage, as well as acceptance of radiation doses that should not be accepted. The access control function shall be maintained throughout the operating period of the facility until the end of the active supervision period (approximately for 300 years).

Risk control measures that can be taken to prevent the failure of access control starting from engineering by placing NSD facilities in a limited environment equipped with security posts, fences, doors, alarms and monitoring cameras. For NSD facility candidates to be built, it has fulfilled several requirements as it is placed in Serpong Nuclear Area (SNA) at PUSPIPTEK, which must at least have to pass 3 (three) layers of guard post before reaching NSD facility. In addition, administrative arrangements (standard procedure) need to be set up for access control and control arrangements in case of nonconformities.

7. Site Characteristic

For Site node, there are 3 (three) main parameters that can be failure factor. First, when viewed from the permeability parameter, SNA site has a higher value than the specified criterion. This will have an impact on the potential for water infiltration to a larger NSD facility. The second, the NSD facility distance to surface water bodies (rivers), the location of the NSD facility is only about 150 meters from the Cisalak River. Not in accordance with the ideal criteria that require the location of NSD facilities to the surface water body of more than 500 meters. This will affect the resistance of natural barrier that is smaller than expected. This means that if there is a release of radionuclides, then the possibility of radionuclides that reach the surface water bodies will be greater. The third is the carrying capacity of the ground. The site where the NSD is built is an area that has rock hardness and less compactness, and low homogeneity. With the condition of such areas, if forced to build facilities disposal demo in the area without engineering it will result in unstable facilities and will experience a rapid decline.

To anticipate the risk due to some characteristic mismatches of the site against the ideal criterion, strengthening is done in the engineered barrier disposal system. The engineered barrier of vault is selected with a low permeability material of reinforced concrete K350 with a thickness of 60 cm to ensure that no radionuclide release occurs. To lower the probability of failure of the facility due to being built in areas with low soil carrying capacity, the demos disposal facility will be built in the vadose zone by using deep foundation.

Conclusion

- The NSD Facility has potential hazards, particularly hazards to radiation safety as it may cause environmental pollution due to the release of radionuclides from waste packages stored in the facility. The determinant of releasing the radionuclides to the environment is the reliability of the existing barrier in the disposal facility. To identify the hazards of failure of each barrier, the HAZOPS method is used;
- Hazards arising from NSD facilities, particularly radiation hazards associated with environmental pollution and radiation exposure, are caused by design errors and human intrusions that may cause deviations from the barrier function. Some parameters of the barrier design that can be used to calculate and estimate risks during the period of operation and active post-closure surveillance are material permeability, the integrity of the waste package, the drainage flow rate, the carrying capacity of the soil, and the nonconformity in access control;
- The use of HAZOPS studies in this study in addition to estimating future risks, may also be used to provide appropriate design, material, and safeguards recommendations for disposal facilities.

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Annex Tabel : HAZOPS for Design of Near Surface Disposal on Serpong Nuclear Area

No.	Node	Parameter	Guide Word	Deviation	Cause	Impact	Risk Control	Recommendation
1.	Cover	Permeability	High	High permeability	incompatibility of design and material selectionl	Rainfall infiltration	El-Cabril design adapted	Monitoring of rainfall infiltration during active surveillance
2.	Backfill	Permeability	low	Low permeability	selection of inappropriate material composition	increase the probability of contacting the waste package with water	selection of appropriate material composition	
3.	Drainage	Flow rate	nothing	There is no flowrate	the gutter tilt is not perfect	Accumulation of infiltrated water	Gutter with slope 1%	
4.	Waste Matrix	Integrity	lower	Low integrity	improper immobilization process	radionuclides are easy to release	cover, vault, drainage, control pool	Waste Acceptance Criteria
5.	Vault	Dose rate	Higher	Dose rate higher than dose constraint	incompatibility of radioactive waste exposure with material type and wall thickness of vault	Individual dose increase	selection and calculation of material thickness appropriate for vault material, WAC	vault using K350 reinforced concrete 60 cm thick
		Permeability	High	High permeability	material selection error vault, erosion, porous	Water infiltration to facility	selection of vault materials according to site characteristics in geological and climatological aspects	vault using K350 reinforced concrete 60 cm thick, periodic monitoring of control pool as an early warning
6.	Control Access	Access control incompatibility	occour	there is undue access	insufficient access control	intrusion, reception of radiation doses should not be	post guard, fence, alarm, cctv, lock settings	
7.	Site	Permeability	Higher	Higher permeability	availability of area	There is infiltration potential and radionuclide release	vault with K350 concrete with 60 cm thickness	
		Distance facility to water body	less	Distance facility to water body < 500 m	availability of area	natural barrier weak	*vault with K350 concrete with 60 cm thickness*	
		Soil bearing capacity	less	The carrying capacity of the soil is not ideal	less rock hardness, less compactness, homogeneity	unstable facility, decreased faster	deep foundation of the vadose zone	

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Extending the Borehole Disposal Experience for the Development of the National Low Level Repository Program

1.0 Introduction

National programme on the development of near surface repository in Malaysia was reviewed in 2016 and the program was strengthened by identifying gaps in the existing program. Among the gaps identified are the lack of experience from the significant number of new staffs and the lack definition of the boundary conditions of the repository. Thus, in order to move forward, the project has identified priorities areas namely:

i. Priority 1 – Inventory. The full inventory of LLW (and possibly ILW) to be disposed of in the national repository need to be defined as well as the final waste form.
ii. Priority 2 - Safety assessment, Safety case and Repository design. For a start, Nuclear Malaysia is planning to undertake a generic safety assessment, in order to initiate the training programme and developing tacit knowledge in the respective fields.
iii. Priority 3 - Stakeholder engagement and communication. Normally, this involves a long process to talk to the public, to enhance understanding and to obtain acceptance. However, this process need to start as early as possible, using all available medium.
iv. Priority 4 – Siting campaign. Full set of the siting process is activated that includes selection of potential site, candidate site and site characterization. The siting process begins with the site screening, of which Nuclear Malaysia has already completed the site screening in 2011.

At this point of writing, Nuclear Malaysia is preparing to begin Priority 1 and Priority 2 concurrently, taking advantage the experience from the Borehole Disposal System Project which specifically focusing on the disposal of Category 3 to Category 5 disused sealed radioactive sources.

2.0 Experience from the Borehole Disposal System Project

2.1 What is a Borehole Disposal System

Malaysia is planning to construct a Borehole Disposal System (BDS) for the disposal of Disused Sealed Radioactive Sources. It is a specially engineered drilled boreholes, with an engineered barrier system, based on the use of stainless steel capsules and buries in a borehole at depths greater than depth 30 meters of that near surface disposal facility. The BDS falls between the two well-established options of disposal in near surface facilities and disposal in geological facilities.

2.2 Project Activities

As a type of a radioactive waste disposal facility, the overall development and implementation of a BDS project constitutes five phases which are the planning phase, design phase, construction phase, operational phase and post closure phase as shown in Figure 1.

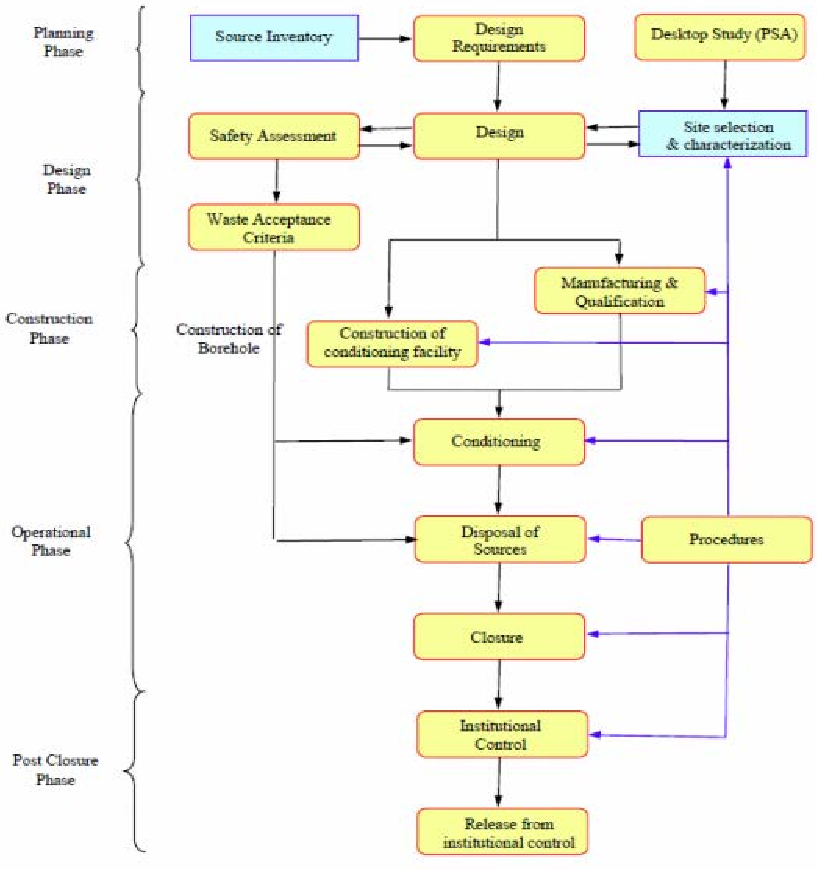

Figure 1: Implementation activities for the borehole disposal facility

During the planning phase, project activities that are being undertaken involve updating the source inventory, identifying design requirements and performing a desktop preliminary study. Activities performed at the pre-disposal phase includes site selection, site characterization, waste acceptance criteria established, waste conditioning that involve a series of source retrieval, encapsulation and containerization, site-specific design of the BDS, safety assessment and safety case, license application, manufacturing and qualification of equipment and last but not least, is the construction of the Borehole facility.

Operational phase or also referred to as the disposal phase constitutes disposal activities like transport and transfer of filled and sealed disposal containers to the borehole, associated backfill of the borehole and finally closure of the borehole. Right after closure, the BDF enters the post closure phase and activities dedicated during this phase are institutional control and eventual release from institutional control.

2.3 Site selection and characterization

The siting guidelines for the BDF system is much less stringent than those of the near-surface and deep geological disposal system as the BDF design provides for the radiological containment in a wide variety of geological environment.

The site characterization activities were conducted for 2 years from 2014 until 2016. The main objectives of the site characterization activities were to observe, inventory, and document the presence, type, location, and relative amount of geological, hydrogeological, geotechnical properties and hydrochemistry. Geological mapping has been carried-out with technical assistance of Minerals and Geoscience Department Malaysia.

Two exploratory boreholes were constructed to determine the actual thickness of the residual soil cover, the actual depth of the contact among the metasedimentary rock underneath as well as to delineate any subsurface geological structural features that intersect the proposed disposal borehole. The depth of the exploratory boreholes exceeded 200 m below the existing ground level. Core logging, geomechanic properties (such as mineral density, bulk density, bearing capacity), hydrogeological testing (such as permeability/hydraulic conductivity, groundwater flow, recharge/discharge) and groundwater sampling for geochemistry testing as well as down hole geophysical logging have been carried out during the exploratory boreholes drilling and after the completion of the drilling work.

The full set of site characterization study covers the assessment of local geology, regional geology, seismic analysis, topography, hydrology, surface erosion, hydrogeology and hydrochemistry. Most of the data that are obtained from the site characterization are used to construct the conceptual models on geological, hydrogeological and geochemical conceptual models. This geological conceptual model demonstrates the geological condition of the proposed site including the geomorphologic, bedrock condition (lithological, deformation zone, fractured, others structural of the soil/rock). It provides the geometrical framework and the geoscientific descriptions necessary for development of the rock mechanics models, hydrogeological models and (hydro) geochemical models, which are required for simulating repository behaviour on the short and on the long term, i.e. for the design and construction of the repository and for the demonstration of safety.

2.4 Post-closure safety assessment and the Safety Case

The post-closure safety assessment is following the Long Term Safety Assessment Methodologies for Near Surface Radioactive Waste Disposal Facilities approach, or in short, the ISAM Safety Assessment Approach. The approach consists of the following key steps:

- The specification of the assessment context;
- The description of the disposal systems;
- The development and justification of scenarios;
- The formulation and implementation of models;

Quantitative analysis was derived using AMBER calculational code. In addition, deterministic approach was employed to assess the sensitivity of parameters used in the conceptual model.

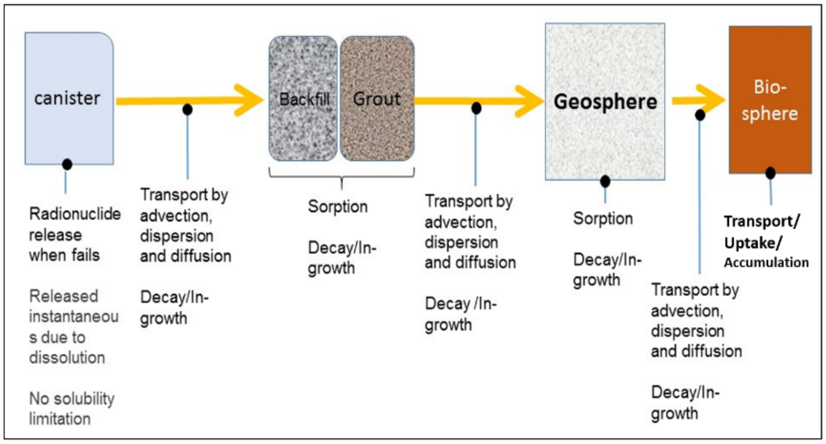

Figure 2: Conceptual model of radionuclide release and migration processes in BDS

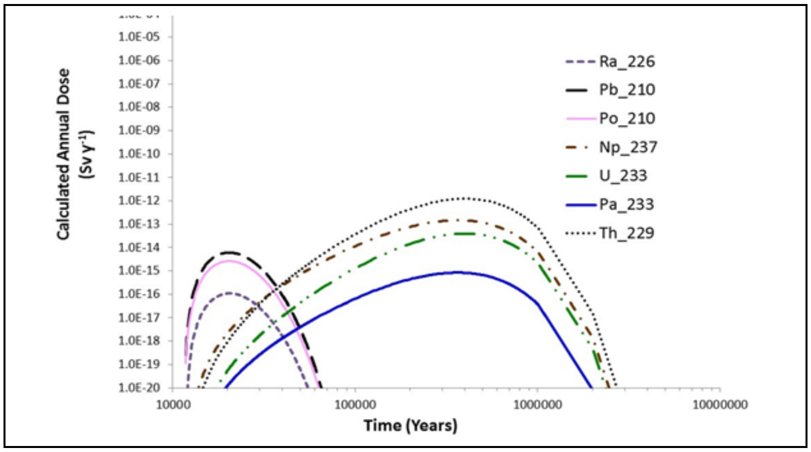

Figure 3: Example of calculated annual dose profile in the post-closure safety assessment

2.5 Project Management

In order to implement the project structurally and efficiently, a project management team was formed that comprises 7 working groups, namely site selection and characterization group, borehole construction and implementation group, safety assessment and safety case group, waste characterization group, waste conditioning group, licensing and research and development group. The project management structure is shown in Figure 4.

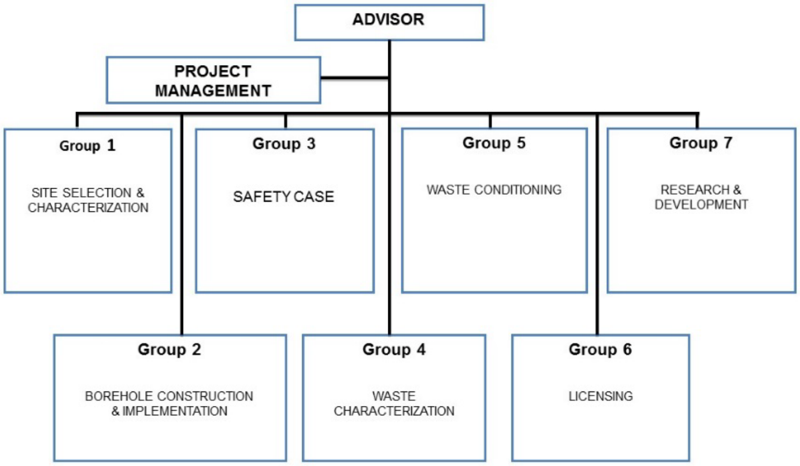

Figure 4 : Project Management Structure

Altogether, there are 30 core personnel in this project comprising of officers and supporting personnel from several divisions and units in Nuklear Malaysia. In addition to strengthening the technical capacity and capabilities, collaboration is made between Nuklear Malaysia (Ministry of Science, Technology and Innovation) and the Ministry of Environment and Natural Resources, where 2 professional geologists from the Mineral and Geological Department, were attached in this project in the site selection and characterization group.

3.0 Extending the BDS Experience for the Near Surface Repository Project

A near surface repository is a large scale project encompasses elements of siting, safety case, design, stakeholders communication and etc. For such project that involves long timeframe, it requires sustainable supporting system inter alia strong project management, efficient knowledge management, and properly planned training of staffs Clearly, the experience gathered during the borehole project is valuable in partly accelerating the technical aspects know-how in particularly with respect to site characterization and post-closure safety assessment.
The training required to support the repository development should cover the overall depth and breadth of a repository project. Training and human capital development should be properly planned and been given an emphasis. Recent training under the Nuclear Researchers Exchange Program, under the purview of MEXT, Japan were carried out on researches related to repository development. The name of the trainee, title of the research project and the year conducted are given below:

i. Azmi Ibrahim – Investigation of Biosphere Model in Safety Assessment of Radioactive waste disposal, University of Fukui, Japan. Year 2016

ii. Kang Wee Siang – A dynamic model approach to simulate current practice of radioactive waste management in Malaysia, University of Fukui, Japan. Year 2017

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Radio-chemical Characteristics of Liquid Radioactive Wastes

Pre-treatment of the liquid waste is the initial step in waste management that occurs after waste generation. It consists of collection, chemical adjustment and decontamination and may include a period of operational (temporary) storage. This initial step provides the best opportunity to segregate waste streams at source. This may facilitate recycling within the process or disposal as non-radioactive waste when the quantities of present radionuclides and their exemption or clearance levels are known. It also provides the opportunity to segregate radioactive waste for different disposal routes. Prior to treatment of radioactive waste it is essential to characterize the radioactive waste accurately. Waste characterization is the determination of the radiological, chemical and physical properties of waste to establish the need for treatment, handling, processing, storage, or disposal of radioactive materials. Typically, characterization is helpful in assessing what must be done to meet the requirements regarding transportation and disposal of radioactive waste. Radiological waste characterization involves quantifying and detecting the radiation characteristics of the principal radionuclides originated as a result from the use of radioisotopes or the production of radionuclides in radiation and nuclear facilities. The active liquid waste collected from research reactor and radioisotope production and other research institution in Bangladesh are generally collect for safe storage at Central Waste Processing and Storage Facility (CWPSF). The present work was conducted as a part of characterization of radioactive wastes which are safely stored at waste storage facility. The liquid waste were analysed for the assessment of the radioactivity concentration.

The Liquid waste which were collected from different users are now safely stored in the Central Waste Processing and Storage Facility. The amount of liquid radioactive waste generated from different users during 2008-20015 is shown in Fig1. The liquid wastes are safely stored at CWPSF in different plastic containers. After collection of effluents the radionuclide contents analysis and measurements of pH and the volume are performed in the spectrometric laboratory.

The active liquid waste collected from research reactor and radioisotope production laboratory was analysed for gamma activity using a 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.

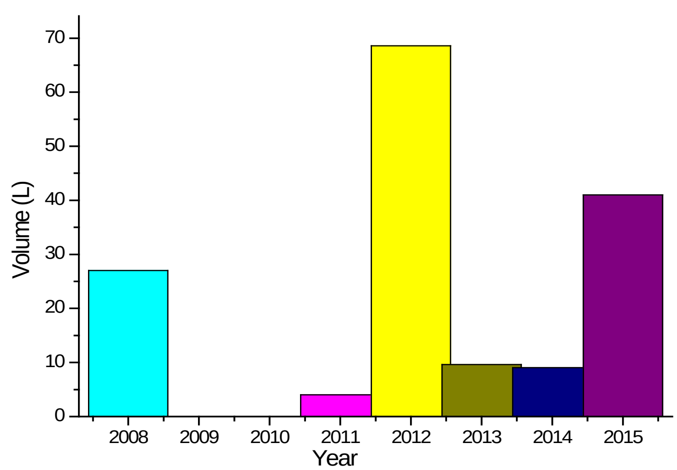


Fig1. The amount of liquid radioactive waste generation

Experimental Methods

To aid risk assessment a non-active run was initially performed. Contamination was checked before collecting the sample from waste container. Latex gloves and leather gloves were worn during the collection of liquid radioactive sample. In order to analyse the liquid radioactive sample, one representative sample from each of the liquid waste container was taken. Protective clothing, lab coats and masks were worn before entering the CWPS F in order to collect the liquid waste sample. The maximum radiation dose rate and surface activity of the liquid waste samples was recorded using an appropriate survey meter. A plastic funnel was used to collect the sample from the active liquid waste container to each of the sample container. Each sample was collected in 250 ml plastic pot and plastic tape was put around the neck of sample pot. Safety mask, polythene, latex and leather gloves were used to avoid contamination and safety spectacles were used to protect eyes from liquid active water drop during the collection of samples. A plastic tray was placed at the beneath of sample container during the collection of liquid sample in order to avoid the contamination with liquid radioactive waste in the floor. The volume of each of the sample pot was measured before and after the loading of liquid waste sample. The sample was then put into a lead pot to take up to the counting room.

The liquid active sample was analysed for γ activity by using a HPGe detector coupled with Genie 200. In order to calculate the activity of the each liquid waste sample, the energy and efficiency calibration was done using Eu-152 standard source. The standard was placed on the top of detector within the shielding arrangement in the counting room of HPRWMU. The spectrometer was set to detect the gamma energy between 149.77 keV to 3578.36 keV. The counting time of the each liquid waste sample was 10,000 sec. The sample was placed in the same position as the standard source for efficiency calibration. The same geometry was chosen for reference standard source for efficiency calibration. A plot of energy versus efficiency is shown in Fig 3. From the sample net counts activity of the each liquid active samples and uncertainty of the activity concentration was calculated using the following standard formula:

$$A = \frac{CPS}{\varepsilon_\gamma \times I_\gamma \times W}$$

where, A is the Activity of the sample in Bq l^{-1}, CPS is the net counts per second, ε_γ is the efficiency of the γ-ray spectrometer at each respective γ-ray energy, I_γ is the intensity of the corresponding γ-ray energy, W is the sample volume.

The uncertainty of the activity:

$$\left(\frac{\Delta A}{A}\right) = \sqrt{\left(\frac{\Delta N}{N}\right)^2 + \left(\frac{\Delta I_\gamma}{I}\right)^2 + \left(\frac{\Delta w}{w}\right)^2 + \left(\frac{\Delta T}{T}\right)^2}$$

where, ΔA is the uncertainty of the sample measurement and ΔN, ΔI_γ, Δw and ΔT are the uncertainties of the count rate, gamma-ray emission probability, sample volume and counting time, respectively.

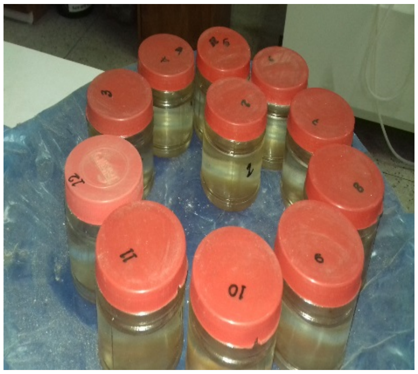


Fig 2. Liquid radioactive samples

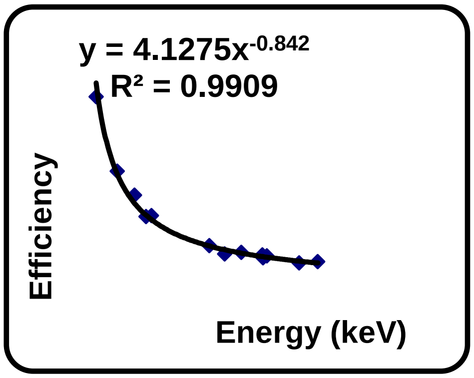


Fig 3. Energy Vs efficiency curve

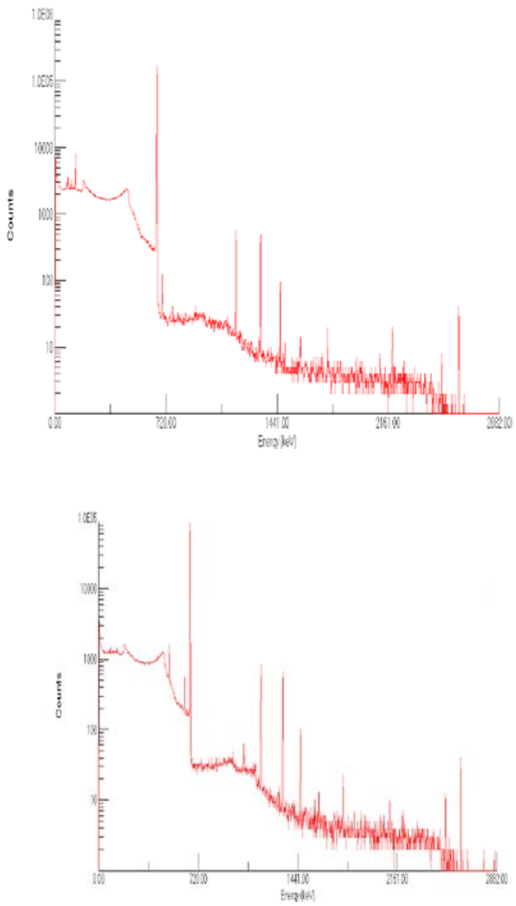


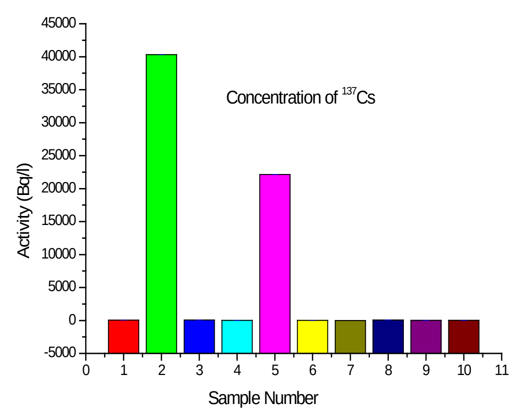
Fig 4. Gamma ray spectra obtained from HPGe detector with 20% relative efficiency

Results

The detection and measurement of radionuclide concentration in the liquid waste samples were obtained using High Purity Germanium (HPGe) detector having 20% relative efficiency. The

Sample	Activity concentration (Bq l^{-1}) of liquid waste			pH
	^{137}Cs (Energy: 661.657 keV)	^{60}Co (Energy: 1173.228 keV)	^{60}Co (Energy: 1332.492 keV)	
1.	30.15±0.24	-	-	5.5
2.	40313.26±0.24	233.12±0.04	235.62±0.06	5.5
3.	85.85 ±0.24	-	-	6.0
4.	8.51±0.34	1335.45±0.03	1314.13±0.06	6.0
5.	22124.93±0.24	362.83±0.03	348.89±0.06	6.0
6.	1.85±0.46	-	-	5.0
7.	-	1755.42±0.03	1726.99±0.06	5.5
8.	70.91±0.24	-	-	5.5
9.	7.26±0.53	3943.82±0.03	3930.03±0.06	5.5
10.	8.96±0.33	2341.26±0.03	2311.26±0.06	5.5
11.	-	113.46±0.04	120.61±0.06	6.5

measured pH of the most of the sample indicates weak acidic nature of the active liquid wastes. The dominant gamma lines found in the spectrum are: 1173.2 keV (0.99), 1332.5 keV (0.99), 661.6 keV (0.85). The identified radionuclides in the active liquid waste samples are ^{60}Co and ^{137}Cs respectively.
The specific activity of radionuclides ^{60}Co, ^{137}Cs in liquid waste samples were analyzed from dominant gamma energy line by using Genie 2000 software. Each of the samples was counted for 10,000 sec and the activities were reported in Bql^{-1} with uncertainty. The activity concentration of ^{137}Cs and ^{60}Co in different liquid waste samples are shown in Fig5. The calculated activity concentration and their respective uncertainties of ^{60}Co & ^{137}Cs are given in Table1.



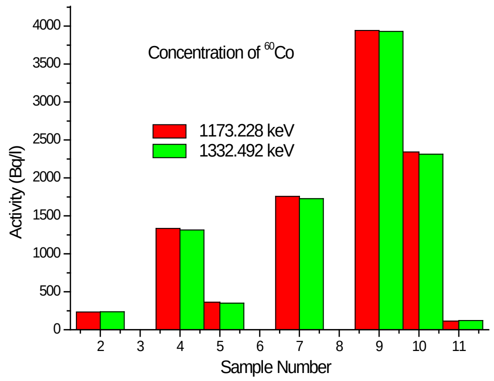


Fig 5. Activity concentration of ^{137}Cs and ^{60}Co in different liquid waste samples

Table1. Calculated activity concentration of ^{137}Cs and ^{60}Co in different liquid waste samples

The activity concentrations of ^{137}Cs in liquid waste samples were found to be varied from (7.26 ± 0.537) Bql^{-1} to (40313.26 ± 0.24) Bql^{-1} whilst the activity concentrations of ^{60}Co having 1173.228 keV and 1332.492 keV gamma lines are ranging from (113.46 ± 0.04) Bql^{-1} to (3943.82 ± 0.03) Bql^{-1} and (120.61 ± 0.06) Bql^{-1} to (3930.03 ± 0.06) Bql^{-1} respectively. The information obtained in this work could be used for the treatment strategy and discharge of the decontaminated effluents under authorization.

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Establishment of New Nuclear Energy for Peace Act B.E 2559 (2016) in Thailand

Introduction

Thailand has used radioactive materials in the country since before 1961. In the beginning. Thailand has no control over this. Thailand enacted the first Atomic Energy Act for peace in 1961. The utilization of the first nuclear research reactor in the country started in 1962. Later, the Atomic Energy for Peace Act was amended in 1965. The use of atomic energy purposes was a necessity for the development of the country. It is important to promote the existence of laws and regulations on nuclear safety and radiation safety, as Thailand is a member country of the International Atomic Energy Agency. (IAEA). The aim of this Atomic Energy for Peace Act 1961 was to control the safety aspects of nuclear research reactor and ionizing radiation.
Since the Atomic Energy for Peace Act 1961 is not present so it is need to be updated.
In accordance with the international standards for the control and prevention of radiation hazards, so the Nuclear Energy Act B.E.2559 was enacted in 2016. In line with modern nuclear technology, which has developed and utilized in the country and to comply with international rules and regulations, including nuclear and radiation treaties, such as nuclear oversight, covering safety, security and safeguard as well as nuclear facilities. The new Act has been in force since February 2017, in order to regulate radioactive material, nuclear material, radiological generator, including radioactive waste. The new Act will promote the safety, security and safeguard in nuclear application in Thailand.

Main important points of Nuclear Energy for Peace Act B.E.2559

In order to use of radioactive material, nuclear material and radiation safely, importers, exporters, manufacturers, users and other stakeholders need to know and comply with the rules and regulations. In this New Nuclear Energy for Peace Act B.E.2559**,** the main issues are:

- Regulation on Radioactive Material
- Regulation on Radiological Generators
- Regulation on Nuclear Material
- Regulation on Nuclear Facilities

Nuclear Energy for Peace Act B.E.2559 has been publishing in the Royal Gazette Vol. 133, No. 67 A, dated 5 August 2016, as shown in Figure 1. The Nuclear Energy for Peace Act B.E.2559 has been in force since February 1, 2017. The Ministerial Regulations , Nuclear Energy Commission Orders and Office of Atoms for Peace Rules under this Act shall be in place after 270 days after the enforcement of this Nuclear Energy for Peace Act B.E.2559.


พระราชบัญญัติ
พลังงานนิวเคลียร์เพื่อสันติ
พ.ศ. ๒๕๕๙
ภูมิพลอดุลยเดช ป.ร.

Fig.1 Nuclear Energy for Peace Act B.E. 2559 [1]

Summary of Nuclear Energy for Peace Act B.E.2559

There are 14 Chapters 152 Articles in the new Nuclear Energy for Peace B.E.2559 as followings,

Chapter 0. Preamble and Definitions (Art. 1-5)
Chapter 1. General Provision for Nuclear Energy (6-8)
Chapter 2. Nuclear Energy for Peace Commission (9-17)
Chapter 3. Radioactive Materials and Radiation Generating Machine (18-35)
Chapter 4. Nuclear Materials (36-44)
Chapter 5. Nuclear Facilities (45-64)
Chapter 6. Radioactive Waste (75-83)
Chapter 7. Spent Fuel (84-87)
Chapter 8. Safety, Security and Safeguard (88-97)
Chapter 9. Transportation (98-99)
Chapter 10. Radiation and Nuclear Emergency (100-102)
Chapter 11. License Revocation (103-104)
Chapter 12. Appeal (105-106)
Chapter 13. Officer (107-114)
Chapter 14. Punishment (115-144)
Chapter Transitory Provision (145-152)

Radioactive waste management and spent nuclear fuel management are in the chaper 6 and 7 respectively.

Chapter 6. Radioactive Waste Management (RWM)

- Import of Radioactive Waste (RW) is prohibited, except re-import of reprocessing of Spent Fuel and conditioning of RW.
- Export/Import of RW should be authorized by OAP Secretary General. Requirement and condition specified in Ministerial Regulations.
- Discharge the RW into the environment is prohibited, except the RW which radio activities and half-life are specified in Ministerial Regulation.
- RW producer must be responsible the management of RW. Requirement and condition specified in Ministerial Regulations.
- RWM service provider shall be got the licenses for siting, construction, commissioning of RWM Facilities and Activities.
- DRSR which more than 5 year no-longer use shall be treated in the manner of Radioactive Waste

Chapter. 7 Spent Nuclear Fuel (SNF)

- Export and Import of SNF are prohibited, except re-import of reprocessing of SNF.
- Export/Import of RW should be authorized by SG. Requirement and condition specified in Ministerial Regulation.
- SNF shall be storaged as specified in Final Safety Analysis Report. Requirement and condition specified in Ministerial Regulation.
- Nuclear facilities have to store their own SNF safely and securely as described in their safety analysis reports (SARs) unless SF is
  - transferred to government entity responsible for SF storage, or
  - exported to be reprocessed, or
  - exported (sent back) to manufacturers or lessors

Reference

[1] Nuclear Energy for Peace Act B.E. 2559, the Royal Gazette Vol. 133, No. 67 A, dated 5 August 2016,

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