4. Recent Status on Decommissioning/Clearance in Japan

4.1 Concept and Strategy of RWM

4.1.1 Regulation for Waste Management

In Japan, the Atomic Energy Commission (AEC) in Cabinet Office (CAO) makes decision on the basic policy on the waste management of nuclear facilities. The AEC states, in a document of "Long-Term Plan for Research, Development and Utilization of Nuclear energy", that the present generations which receive the benefits of nuclear energy, are obliged to do their utmost to ensure the safe disposal of radioactive waste generated in the research, development and utilization of nuclear energy, and should invest continued efforts in achieving this goal, that the waste producers has the primary responsibility for safe processing and disposal of the waste, and that the government has the responsibility for taking necessary measures to ensure that this processing and disposal are carried out appropriately and safely by the producers, through giving adequate guidance and establishing necessary regulations. Furthermore, the AEC states that the government should play an appropriate role in implementing disposal program for radioactive waste, particularly high level waste (HLW), ensuring long-term safety, in addition to its activities related to promotion of research and development activities and safety regulation.

According to the policy, the Nuclear Safety Commission (NSC) in CAO establishes policy in safety standards such as upper bounds of radioactivity concentration for disposal of radioactive materials and methods for safety assessment of disposal facilities. NSC also plays a role in investigation and examination concerning basic principles for regulation of radioactive waste management, establishment of safety standards of radioactive waste management and review of the safety examinations of radioactive waste management facilities by regulatory bodies. Ministry of Economy, Trade and Industry (METI) and Ministry of Education, Culture, Sports, Science and Technology (MEXT) have established and continued to improve the legal framework.

Regulations of the waste management for different nuclear and radioactive facilities are implemented by METI and MEXT, and Ministry of Land, Infrastructure and Transport (MLIT). The roles of Radioactive Waste Regulation Division, Nuclear and Industrial Safety Agency (NISA) of METI are drafting of regulatory laws and provisions, regulation of radioactive waste disposal facilities and storage facilities, regulation of off-site radioactive waste management, regulation of decommissioning of nuclear facilities. The Radioactive Waste Safety Subcommittee, Nuclear and Industrial Safety Subcommittee, Advisory Committee for Natural Resources and Energy, METI is responsible to investigation into safety policy concerning radioactive waste disposal and storage. The Nuclear Safety Division, Science and Technology Policy Bureau, MEXT is responsible to regulation of management of radioactive waste originating from scientific use of nuclear materials and use of radio-isotopes and radiation-generating apparatuses. The role of Technology and Safety Division, Policy Bureau, MLIT is regulation of the maritime transportation of radioactive waste.

The Framework of legislation system is established under The Atomic Energy Basic Law (The Basic Law) which consists of two categories of laws. One is for "The Law for the Regulation of Nuclear Source Material, Nuclear Fuel Material and Reactors (Reactor Regulation Law)" which was established to ensure the peaceful and safe uses of nuclear source materials, nuclear fuels and nuclear reactors. The other is for " The Law Concerning Prevention from Radiation Hazards due to Radio-isotopes, etc. (Radiation Hazards Prevention Law) which was established for the purpose of regulation of radio-isotopes and radiation-generating apparatuses.

METI and MEXT have established and continued to improve the safety standards consisting of the laws, for safe and proper processing, storage and disposal of radioactive waste, on the bases of studies and decisions made by the AEC and the NSC. The criteria for gaseous and liquid radioactive waste discharge have been established in accordance with relevant international recommendations. Solid radioactive waste is classified into two categories, namely HLW (liquid waste generated from spent fuel reprocessing and its vitrified package) and other LLW. The LLW is sub-classified according to origin (differing radionuclide composition) and level of radioactivity.

The disposal of low-level radioactive waste (LLW) from nuclear reactor operation has been implemented since 1992. For management of HLW, the Specified Radioactive Waste Final Disposal Act was issued by the Diet in 2000. The act stipulates the procedures of site selection, executive body and fund accumulation in relation to the disposal of vitrified waste from the reprocessing of spent fuel. A following three-step site selection procedure is defined; the selection of "Preliminary Survey Site" where deep underground conditions are surveyed by boring or other methods, the selection of "Precise Survey Site" where detail survey with underground facility is carried out, and the selection of a "Final Disposal Site" for building the final disposal facility.

With respect to clearance, AEC decided to introduce a procedure of clearance for waste from nuclear reactor and NSC considered that the exemption level for bulk materials provided in the IAEA safety guide, RS-G-1.7 can be used for clearance level. According to the decisions, the Reactor Regulation Law was amended in May 2005 to provide for clearance level and the procedure for its verification, while the relevant regulations are going to be established in the future.

Basic concepts for safety regulation of the disposal of the other types of radioactive waste such as transuranic (TRU) waste (originating from reprocessing facilities and MOX fuel fabrication facilities), uranium waste (originating from uranium fabrication facilities) and so on are not yet established.

4.1.2 Low Level Radioactive Waste Management in Japan

1. Introduction

Low level radioactive waste in Japan is arising from commercial nuclear power plants, nuclear fuel cycle facilities such as fuel fabrication plant or reprocessing plant, research facilities such as research reactor or accelerator, and medical applications including radiopharmaceutical production.

According to Framework for Nuclear Energy Policy by Japan Atomic Energy Commission (2006), there are four principles of the treatment and disposal of radioactive waste arising from such a various institutions: 1) the liability of generators, 2) minimization of radioactive waste, 3) rational treatment and disposal, and 4) implementation based on mutual understanding with the public people. Under these principles, it is important to make appropriate classifications of the wastes and treat and dispose of them safely for each classification based on the recognition that the wastes may include materials with characteristics that take an extraordinary long time for the radioactivity to drop to insignificant levels.

2. Disposal concept of low level radioactive waste

Disposal concept of the low level radioactive waste in Japan that is shown in Figure 1 is applied following to the radioactivity concentration; 1) near-surface disposal without engineered barriers (very low level waste), 2) near-surface disposal with engineered barriers (low level radioactive waste), 3) sub-surface disposal with engineered barriers (relatively high βγ radioactive waste). The low level radioactive waste with high concentration transuranium arising from reprocessing plant, MOX fabrication plant is applied to the geological disposal.

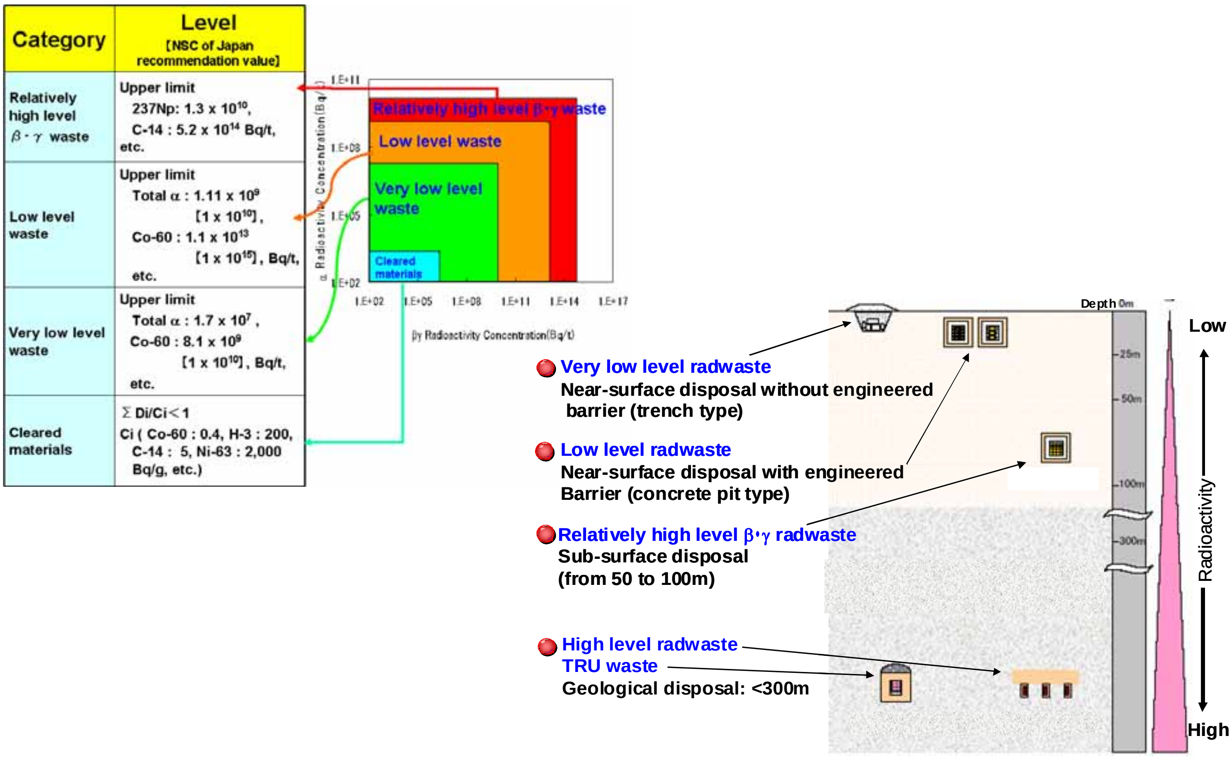

Figure 1 Disposal concept of LLW in Japan

3. Current status of radioactive waste disposal

Japan Nuclear Fuel Limited (JNFL) conducts the low level radioactive waste disposal business with the near surface disposal with engineered barriers at Rokkasho, Aomori Prefecture. The Low Level Waste Disposal Center of JNFL accepts the low level radioactive waste generated in commercial nuclear power plants. The center has now its No.1 Disposal Facility in operation with a capacity of 40,000m^3(homogeneous waste (solidified liquid waste with cement, bitumen or plastic into 200-liter drums), operation started on 1992) and No.2 Disposal Facility (40,000m^3, heterogeneous waste (solidified solid waste with cement into 200-liter drums), operation started on 2000.

Near-surface disposal without engineered barriers is conducted partly while operation entities improve safety regulation systems on the remaining part of the wastes. Japan Atomic Energy Agency (JAEA) disposed of the approximately 1,670 tons concrete waste generating from dismantling of research reactor of JPDR on 1995 at Tokai, Ibaraki Prefecture. The radioactive concrete waste was packed into a polyethylene and polyester sack called a flexible container (external dimensions of about 1 m in diameter and height, and a capacity of about 0.8 m^3.) to handle waste safely and to prevent dust generation. The disposal site had already been caped by final cover soil and grass. Now this site is under institutional control for 30 years.

In dealing with waste from research and development facilities, TRU waste and uranium waste, discussions on safety regulations have been undertaken step by step.

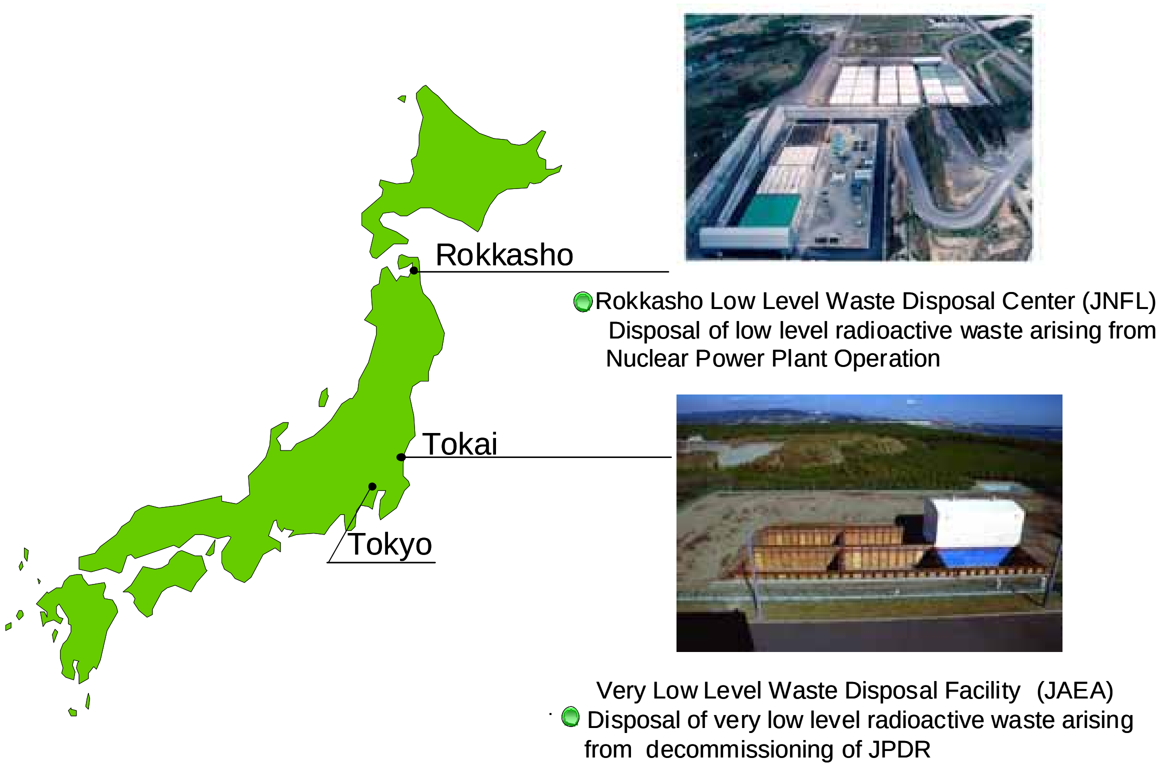

Figure 2 Site location of disposal site in Japan

4.1.3 Clearance

Reactor facility demolition ends up with a huge amount of radioactive waste. The clearance system has merit to decrease the amount of the waste which should be treated as radioactive. Securing public safety is essential to introduce the system.

The “Safety Guide RS-G-1.7: Application of the Concepts of Exclusion, Exemption and Clearance” (hereafter, the IAEA Guide), the dose criterion when the exemption level of the radionuclide of artificial origin is decided is shown the order of 10 μSv/y or less in effective dose to the individual with the radionuclide concerned.

The activity concentration that may be used for exclusion, exemption, and clearance to radionuclide of natural origin is shown in Table 1. that of artificial origin about typical nuclides appears in Table 2.

Table 1. Value of activity concentration for radionuclides of natural origin to applied for exclusion, exemption, and clearance

Radionuclide	Activity concentration (Bq/g)
K-40	10
All other radionuclides existing naturally other than above	1

Table 2. Values of activity concentration for main radionuclides of artificial origin, in bulk

Radionuclide	Activity concentration (Bq/g)	Radionuclide	Activity concentration (Bq/g)
H-3	100	Sb-125	0.1
C-14	1	Cs-137	0.1
Mn-54	0.1	Eu-152	0.1
Co-60	0.1	Pu-239	0.1
Sr-90	1	Am-241	0.1
Tc-99	1	Cm-244	1

Based on the IAEA guide, the Nuclear Safety Commission of Japan reviewed and reevaluated the clearance level of the reports, which were subsequently published by the Nuclear Safety Commission. The calculation value of the clearance level reevaluated is shown in Table 3.

Comparing the calculation value of the clearance level reevaluated with the calculation value of the clearance level in the report, the large difference of the numerical values is to only Ni-59 and Ni-63 of 1/10 less or 10 times more. We can conclude that there was no great divergent value about most nuclides, even though it is different from most nuclides.

Table 3. Minimum value of activity concentration of equivalent to 10μSv/y and calculation value at clearance level in each examination item (Bq/g)

Radionuclide	Calculation value reevaluated at clearance level	Calculation value at clearance level in the previous report
H-3	64	220
C-14	3.8	5.1
Mn-54	1.6	1.1
Co-60	0.31	0.37
Ni-59	35	560
Ni-63	130	1900
Pu-239	0.19	0.20

In the IAEA guide, the standard of the amount of an individual dose when the exemption level is derived should be about 10μSv/y or same level. Moreover, because the exemption level is a very low value in comparison with the radiation level of the natural world that we usually receive, it is admitted that the actual exemption level to be implemented is about 10 times less than the derived exemption level by the judgment of each country's regulatory body.

Based on above, the difference between the reevaluated calculation value of the clearance level shown by the Nuclear Safety commission of Japan and the calculation value of the IAEA guide is approximately the majority of radionuclide within one figure; it is almost equal substantially to the ideas of the IAEA guide.

With regard to the examination of the Nuclear Safety Commission, the Nuclear and Industrial Safety Agency carried out an examination in the Waste Safety Subcommittee on the fundamental matters of necessary technological requirements for clearance verification, assuming a concrete safety regulation for the verification by the regulatory body on the basis of sharing a role between the state and the owners concerning the verification system focusing on the clearance level verification method that is highly reliable and reasonably applicable. The report entitled "Maintenance of the Clearance System in Nuclear Installation" was then published.

The clearance system is already started with the establishment of the verification system. Metal materials coming from dismantling of nuclear reactor facilities are now available to reuse, and they are offered to make benches for the public.

In Japan, there are two laws regarding radiation regulation; “Nuclear Reactor Regulation Law” and “The Law Concerning Prevention of Radiation Hazards”. The former regulates nuclear reactors, and the latter regulates the use of radioisotopes in medical institutions and research organizations. Clearance system concerning the nuclear reactors runs steadily, but that regarding radioisotope wastes still has many problems to overcome. Radioisotope wastes are brought from various fields. Over 200 kinds of radioactive nuclides are used in more than 2000 facilities. This complexity makes the establishment of verification system difficult. How to establish a beneficial clearance system of radioisotope wastes is now being discussed.

4.2 Case Study on Decommissioning and Disposal

4.2.1 Research Reactor Decommissioning

Since starting the research and development of nuclear energy for peaceful use in 1956 in Japan, various research and demonstration facilities have been constructed in research organizations, universities and commercial sectors. Some of the nuclear facilities constructed in the early stage of research and development have been retired to be decommissioned because of completion of their initial objectives. The Japan Power Demonstration Reactor (JPDR) has been decommissioned to recover green field conditions and the decommissioning programs of some research rectors, the reprocessing test facility, and fuel cycle facilities are on going or in preparation in Japan.

Table 4.1 shows status of research reactors that are in operation or under decommissioning in Japan, where fifteen reactors are in operation and eight are under decommissioning. The followings are the outline of two decommissioning projects, the JPDR and Japan Research Reactor No.2 (JRR-2) projects.

Table 4.1 Research reactors in Japan

Facility	Owner	Thermal Power	Status
Kinki UTR	Kinki Univ.	1 W	operation
KUR	Kyoto Univ.	5 MW	operation
KUCA	Kyoto Univ.	100W	operation
Yayoi	Tokyo Univ.	2 kW	operation
YNCA	Toshiba	200 W	operation
STACY	JAEA	200 W	operation
TRACY	JAEA	10 kW	operation
JRR-3	JAEA	20 MW	operation
JRR-4	JAEA	3.5MW	operation
NSRR	JAEA	300 kW	operation
FCA	JAEA	2 kW	operation
TCA	JAEA	200 W	operation
JMTR	JAEA	50 MW	operation
HTTR	JAEA	30 MW	operation
Joyo	JAEA	140 MW	operation
Musashi Tech.	Musashi Tech. College	100 kW	decom.
Rikkyo	Rikkyo Univ.	100 kW	decom.
HTR	HEC	100 kW	decom.
TTR-1	Toshiba	100 kW	decom.
JRR-2	JAEA	10 MW	decom.
VHTRC	JAEA	10 W	decom.
Mutsu	JAEA	36 MW	decom.
DCA	JAEA	1 kW	decom.

<u>JPDR decommissioning project</u>

JPDR is a BWR type reactor with the power of 90 MWt, facility operation period is from 1963 to 1976. The JPDR decommissioning project was the first experience of dismantling of reactor facility in Japan. The objectives of this project are to gain actual experience of nuclear power plant dismantling, to verify the developed techniques in actual dismantling activities, and to collect data on JPDR dismantling activities. The project which began in 1981 was successfully completed to recover green field conditions by March 1996. The basis of decommissioning project was that the plan of dismantling activities was made based on the guideline issued by Nuclear Safety Commission[1)], and that safety of workers and prevention of radioactive materials being released into the environment were considered.

The project was divided into two phases; technology development (1981-1986), and actual dismantling (1986-1996). Various technologies for reactor decommissioning were developed in the first phase[2]. The developed technologies were applied to the actual dismantling of the JPDR in the second phase. The reactor pressure vessel was successfully cut by the underwater arc saw cutting system, and the concrete biological shielding was dismantled by abrasive water jet and mechanical cutting. The controlled blasting method was applied to dismantling the concrete biological shield of extremely low radioactivity. Data on project management such as manpower expenditure, waste arising, worker dose were systematically collected in the dismantling activities. The data was analyzed to characterize the dismantling activities. The radioactive waste arising from the dismantling activities weighed 3,770 tons, worker dose was 306 mSv, and the project cost was approximately 23 billion Japanese yens including the first and second phases. The data was also analyzed to find out unit productivity factors that are the relationships between manpower expenditure and certain indexes such as weight of components dismantled, work areas and capability of dismantling machines[3]. The results were applied to development of arithmetic models and unit activity factors for forecasting of manpower need in other dismantling projects. Figure 4.1 shows example of unit productivity factors for various components dismantled.

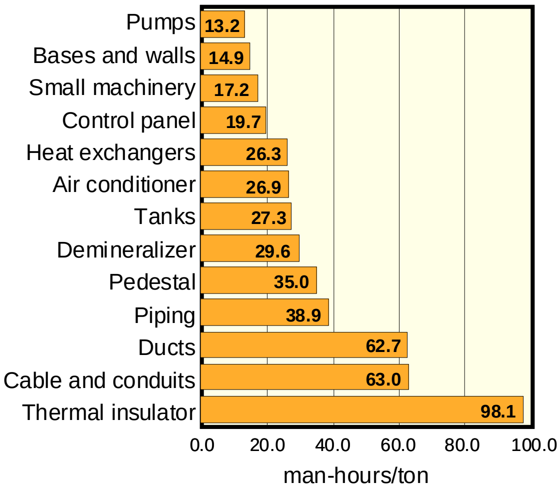

Fig. 4.1 Unit productivity factor

Main lessons learned from JPDR dismantling activities are summarized as follows:

- Radiation confinement systems such as air blowing/exhausting system, under water cutting, local ventilation system were effective.
- Mockup test for pre-evaluation of dismantling activities was useful for dismantling procedures, handling sequences, and time evaluation.
- It was recognized that wide-use remote handling system was necessary to minimize amount of waste and to reduce time for installation.

JRR-2 decommissioning project[4]

JRR-2 is a heavy water moderated and cooled research reactor. It attained criticality in October, 1960 and continued operation until 1984 for neutron scattering experiments, irradiation tests of nuclear fuels and materials, radioisotope production, boron neutron capture therapy, etc. The JRR-2 was finally shut down due to degradation of facilities after 36 years of operations. The JRR-2 decommissioning project started in August, 1997. The project was divided into 4 major phases. In Phase 1 (1997), reactor was permanently shutdown, where function of reactor was lost, light water and heavy water were extracted. In Phase 2 (1998-1999), cooling system and reactor safety storage were isolated, and extracted heavy water was shipped to Canada. In this stage, reactor body itself was sealed up. In Phase 3 (2000-2002), decontaminating and dismantling of dismantling of cooling system were carried out, where heavy water tank, main heat exchanger and main pump were dismantled. The worker exposure was well lower than the expected value in Phases 1, 2 and 3. Now the reactor body is under safe storage before Phase 4 where the reactor body will be removed by one piece removal or one piece caisson technique.

References

1) Nuclear Safety Commission: Basic Concepts for Assuring Safety in Dismantlement of Nuclear Reactor Facility, (1985) [in Japanese].
2) M. Ishikawa, et al.: JPDR Decommissioning Program –Plan and Experience-, Nucl. Eng. & Des., Vol. 122, 357(1990).
3) T. Sukegawa, et al.: Development of Project Management Data Calculation Models Relating to Dismantling of Nuclear Facilities, JAERI-Data/Code 99-005 (1999) [in Japanese].
4) T. Suzuki, et al.: JRR-2 Decommissioning Activity (2), JAERI-Tech 2005-018 (2005) [in Japanese].

4.2.2 Medical Facility Decommissioning

The greatly different points about the decommissioning of medical facilities compared with reactor facilities are that;

・the scale of a facility is small.

・there is no radioactivation or a small range.

・a clearance system is not legislated.

・the authorization of decommissioning plans is not needed. (The relation with the regulation body is only the notification of discontinuation and the reports of measures taken following discontinuation after the discontinuation.)

Radiation facilities of medical field are divided into three types generally, so I introduce the situation of the decommissioning measures about each.

1. Radiation exposure facilities by sealed radioactive sources
2. Nuclear medicine facilities with unsealed RI (radioisotope)
3. Radiation exposure facilities by radiation generators

1. Radiation exposure facilities by sealed radioactive sources

Only use room is appointed in the controlled area in almost facilities of this type.

Because RI using is device equipped with sealed radioactive source, the facilities do not have exhaust and drainage systems, and usually are not contaminated. Therefore radioactive wastes do not occur under using and decommissioning.

The decommissioning is finished by the release of the controlled area, after return of sealed radioactive sources to the distribution source and confirmations of non-contamination by the survey of the wall and the floor in the controlled area

The rooms at the facilities after discontinuation does not have any contaminations so can be used for other purposes.

2. Nuclear medicine facilities with unsealed RI

In the facilities of this type RI is used for the purpose of radiopharmaceutical therapy or diagnostic imaging administrated to the human body.

In many cases this RI is liquid unsealed, the facilities have the contamination test room, the storeroom, and the storing disposal facility other than the administration room and the camera room. Furthermore, the facilities have the drainage and exhaust systems, HEPA filters box, drainage cleanup tanks, etc. as disposal equipments.

A number of 10 nuclides are used as radio-pharmaceutical, each half-lives are shorter than 60 days. However, in the case of discontinuation the confirmations of non-contamination by the surveys of the whole facility are needed. Because in Japan the wastes judged contaminated with RI once is treated as radioactive wastes even if it passes how many years, unless the contaminated part is removed and non-contamination is confirmed.

It is usually judged that there is not contamination if a survey result is less than 3 times of the standard deviation of BG.

Even if there is contamination it can be removed relatively easily by wiping it off because it is only surface contamination by the known nuclides.

The radioactive wastes occurred with the discontinuation of the radiation facilities must be entrusted with waste disposal to a permitted disposer.

If contamination is removed the work rooms can be used for other purposes like 1. after having removed the controlled area, too.

3. Radiation exposure facilities by radiation generators

In the facilities of this type, there is a radiation generator, and a permitted user irradiates accelerated particles in the affected part of the patient or produces radiopharmaceutical such as FDG in a hospital.

When the output energy of a generator is low, a facility is not radioactivated, and only the target of a generator and the penumbra are treated as radioactivated materials.

On the other hand, in facilities with high output energy, the facilities are radioactivated like a case of the nuclear reactor. Therefore the radioactivated concrete husk other than the body of a generator occurs as radioactive wastes.

Nuclides and radioactivity of the wastes are divided into small blocks and taken the samples, then they are measured. In the case of the concrete of the wall and the floor, they are measured in a form of core samples. It is usually judged that there is not radioactivated if a measurement result is less than 3 times of the standard deviation of BG.

When radioactivated part of the concrete is near surface, only the part judged radioactive is cut out and becomes radioactive wastes.

If contamination and radioactivated parts are removed the generator rooms can be used for other purposes like 1.and 2. after having removed the controlled area, too.

However, when radioactivated parts are deep, the facilities are scrapped in many cases for reason of that there is not needs of the reuse or the recycling is difficult.

The radioactive wastes occurred with the discontinuation of the facilities must be entrusted with waste disposal to a permitted disposer.

In the current Japanese system, the radioactive wastes occurred in medical facilities is delivered to a permitted disposer, and a permitted disposer must store them forever under management or deliver to another permitted disposer, because there is not final disposal system and clearance system.

However, it is entrusted with whether a waste is a radioactive waste or not to the judgment of the person in charge of medical facilities, therefore the need of the confirmation by the third person in clearance system does not occur.

The clearance system is regarded as a rational method of the radioactive waste, but even if the system is established there are few merits for most radiation facilities, and the system is still in a talking stage. Because the amount of materials of radioactive wastes occurred in radiation facilities as well as medical facilities is small including radioactivated materials.

4.2.3 TENORM Waste Disposal

1. TENORM Waste

Japan is dependent on import from overseas for raw materials used for the industrial activities. The annual amount of those raw materials that are most imported exceeds 100 million tons.

The survey on the minerals for current industrial use covers monazite, phosphate ore, titanium minerals, bastnaesite, zircon, coal and samarium oxide.

Using 1Bq/g, the BSS exemption level of Th-232 and U-238 series, the survey identified minerals that exceeded this reference criteria in samples taken from the plants handling monazite, phosphate ore, zircon and bastnaesite. The waste arising from these minerals in the separation processes is normally treated as industrial waste, however, the activity concentration in the waste that exceeded the reference criteria was not found.

2. Current Regulation of NORM

The Law concerning Prevention of Radiation Hazards due to Radioisotopes, etc (hereafter referred to as the "Radiation Hazards Prevention Law") and the Nuclear Reactor Regulation Law regulate radioactive substances based on their activity concentrations and activities.

The Radiation Hazards Prevention Law was amended in June 2005, which introduced the exemption level based on the BSS, except for nuclear materials.

The Nuclear Reactor Regulation Law specifies that nuclear materials shall be controlled if their activity concentrations exceed 74 Bq/g and 370 Bq/g in case of solid nuclear materials. Also the Nuclear Reactor Regulation Law specifies that it shall be obliged to notify the use of nuclear materials of which total weights of the tripled weights of uranium plus the weight of thorium exceed 900g.

3. Classification of NORM-containing Substances

The General Administrative Group of the Radiation Review Council proposed the classification of NORM-containing substances and the measures for individual regulation. The proposed categories are as follows;

Category 1 : Substances that are not treated to increase a ratio of NORM-containing minerals and ores (Excluding Categories 2, 3, 4, 5, 6)

Category 2 : NORM-containing residues that were disposed of in the past

Category 3 : Ash and scale generated from industries (raw material substances whose concentration is below an exemption level)

Category 4 : Surplus soil form operating mines, residues after industrial use (disposal)

Category 5 : Industrial raw materials (fabrication, energy production, mining)(excluding Category 7)

Category 6 : Consumer goods (use)

Category 7 : Nuclear fuel substances and radium sources that were refined for radiation use and substances used as radiation sources

Category 8 : Radon

4. Disposal of TENORM waste

Disposal of TENORM waste is mainly concerned with Categories 2 to 5. TENORM waste may be generated from Categories 2 to 5, however, there is no specific regulation for the disposal of TENORM waste. Only the Mine Safety Law specifies the dose limit (1mSv/year) outside the supervised area of the mine.

Figure 4.2.3-1 shows the disposal concept of uranium mill tailings.

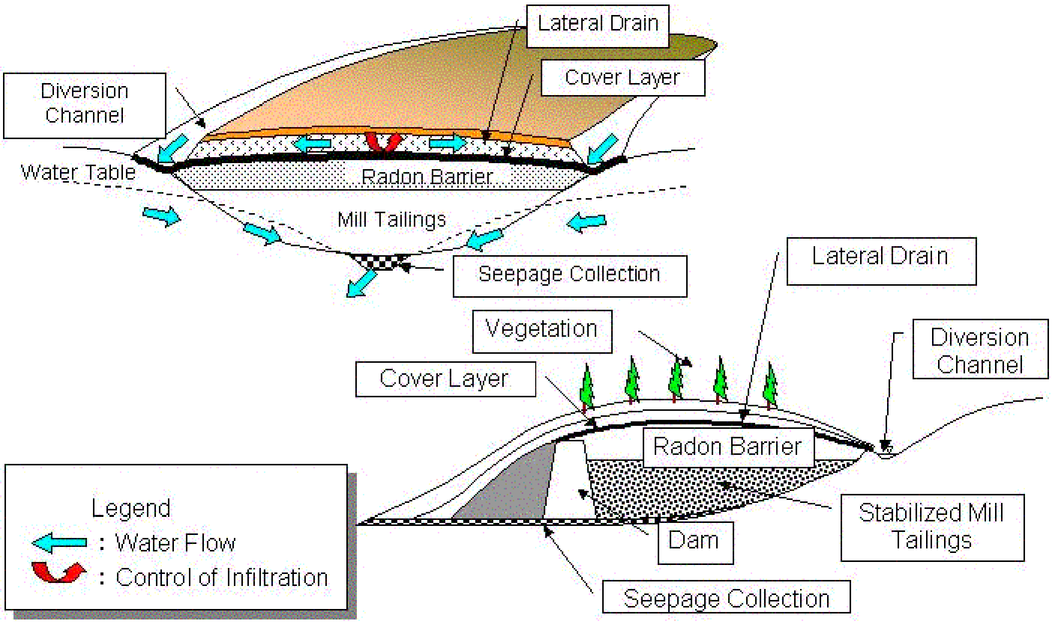

Figure 4.2.3-1 Disposal concept of uranium mill tailings

Other wastes generated form industrial activities are normally treated and disposed as industrial waste.

The disposal of industrial waste is regulated by the Waste Management and Public Cleansing Law. This law specifies the final disposal of industrial waste as stabilized type, controlled type and isolated type.

As the TENORM waste has not yet regulated except for waste rock residues and soil from mines which are regulated under the mine safety law, almost TENORM wastes are seemed to be disposed as industrial waste into controlled type industrial disposal facility. Further study and investigation are expected to identify the TENORM waste and to manage or dispose them.

4.3 Problem to be solved

The first application for clearance for metal waste generated in decommissioning the Tokai Power Station of the Japan Atomic Power Company was approved on May 2007. In future, concrete waste will be also a target for clearance application, but it has the specific characteristic that natural radioactive nuclides, such as K-40, uranium series and thorium series, are contained in the concrete materials. The main radioactive nuclide in solid materials generated from nuclear power plants is Co-60, which is a gamma ray emitter. It is necessary to subtract the contributions of natural radioactive nuclides from measurement results in order to judge whether clearance criterion is satisfied using gamma ray measurement. Usually, it is easy to determine the contributions of natural radioactive nuclides by gamma spectroscopy with a Ge semiconductor detector. However, the Ge detector is not suitable for clearance measurements since its detection efficiency for gamma rays is considerably lower than that of a large plastic scintillation detector. It is also difficult to correct the self-shielding effect of the measurement target. This is the main reason why the large plastic scintillation detector is frequently used for clearance measurements irrespective of its inadequacy for gamma spectroscopy. This situation leads to the necessity of obtaining net count rates for measurement target nuclides, such as Co-60, by appropriately subtracting the contributions of natural gamma radiation from gross count rates. It should be noted that the activities of target nuclides would be underestimated for clearance if the contributions of natural gamma rays are overestimated. To enable the clearance of concrete waste, one of the issues to be solved is making a guideline for the appropriate estimation of the contribution of natural gamma radiation.

In addition, there is also the difficult problem of how to treat the uncertainties of the decay of easily measurable Co-60 resulting from unclear historical records on past contamination when we carry out the clearance of metal and concrete wastes that have been packaged in drum containers and kept in a solid waste storage facility. In such cases, since clearance judgment is usually based on the measurement results of gamma emitters and estimation results of difficult-to-measure beta or alpha emitters, another important issue is to make a guideline on how to determine beforehand the nuclide spectrum or the nuclide vector, which is the ratio of the concentrations of beta or alpha emitters to that of easily measurable gamma emitters.

Although exemption levels for bulk solid materials in the IAEA safety guide, RS-G-1.7, have been adopted in the clearance levels in Japan, the guide gives an allowance of up to ten times the exemption levels in the RS-G-1.7 when deciding national clearance levels in consideration of the various backgrounds of countries in the world. In Asia, there are various countries where public exposure to natural radiation is quite high or mortality due to non-radiation risk is high in relation to the completeness of medical and hygienic infrastructures. It should be carefully discussed whether or not the exemption levels in the RS-G-1.7 can be adopted as national clearance levels like as Japan. The drawing up of guidelines on how to consider radiation protection regulation reasonably in relation to the various backgrounds of Asian countries should be recognized as an issue to be solved by Japanese experts who are expected to be leaders in the scientific field of radiation protection.

4.4 Conclusion

The concept of clearance in Japan was developed with consideration given to the Japanese situation of radiation protection system referring to those in international authorities such as ICRP, IAEA and European Commission.

Through some experiences of performing dismantling of nuclear/radiation facilities, a lot of valuable data was collected. However, problems to be solved are still remaining.

When a decommissioning of nuclear/radiation facility is carried out using the concept of clearance, the most important thing is adopting technical know-how of experienced countries with modification in the way of the home country's situation. Therefore, relevant knowledge should be actively exchanged among countries with different experiences through this FNCA-RWM activity.